

STRATEGIC ENERGY TECHNOLOGY PLAN

Scientific Assessment in support of the Materials Roadmap enabling Low Carbon Energy Technologies

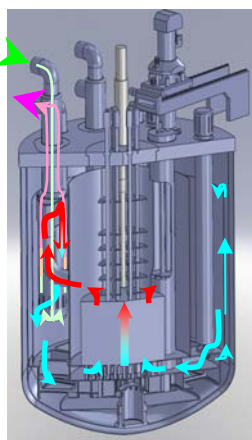
Technology Nuclear Energy

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**Conceptual Design
SFR**



**Conceptual Design
GFR**



The mission of the JRC-IET is to provide support to Community policies related to both nuclear and non-nuclear energy in order to ensure sustainable, secure and efficient energy production, distribution and use.

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Preamble

This scientific assessment serves as the basis for a materials research roadmap for the nuclear fission technology, itself an integral element of an overall "Materials Roadmap Enabling Low Carbon Technologies", a Commission Staff Working Document published in December 2011. The Materials Roadmap aims at contributing to strategic decisions on materials research funding at European and Member State levels and is aligned with the priorities of the Strategic Energy Technology Plan (SET-Plan). It is intended to serve as a guide for developing specific research and development activities in the field of materials for energy applications over the next 10 years.

This report provides an in-depth analysis of the state-of-the-art and future challenges for energy technology-related materials and the needs for research activities to support the development of nuclear fission technology both for the 2020 and the 2050 market horizons.

It has been produced by independent and renowned European materials scientists and energy technology experts, drawn from academia, research institutes and industry, under the coordination the SET-Plan Information System (SETIS), which is managed by the Joint Research Centre (JRC) of the European Commission. The contents were presented and discussed at a dedicated hearing in which a wide pool of stakeholders participated, including representatives of the relevant technology platforms, industry associations and the Joint Programmes of the European Energy Research Associations.

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1. Technology and System State of the Art and Challenges

The development of nuclear reactor technologies (Figure 1.1) has been driven by expansion of nuclear energy, improved safety and economy and sustainability of nuclear systems. Material performance has often been the limiting factor and the development and deployment of materials has always been a key issue. The two main areas for the nuclear materials are fuel and structural materials. The fuel is related to the availability of uranium and the possibility of recycling spent fuel. Structural materials include materials (usually steels) for large and long-life components in particular the reactor vessel but also other components such as core support, steam-generators and intermediate heat exchangers. The fuel assemblies and other reactor pressure internals experience the highest temperatures and irradiation levels. Key functional materials include for instance control rods and pumps. For the nuclear safety concrete structures are also important as they provide containment.

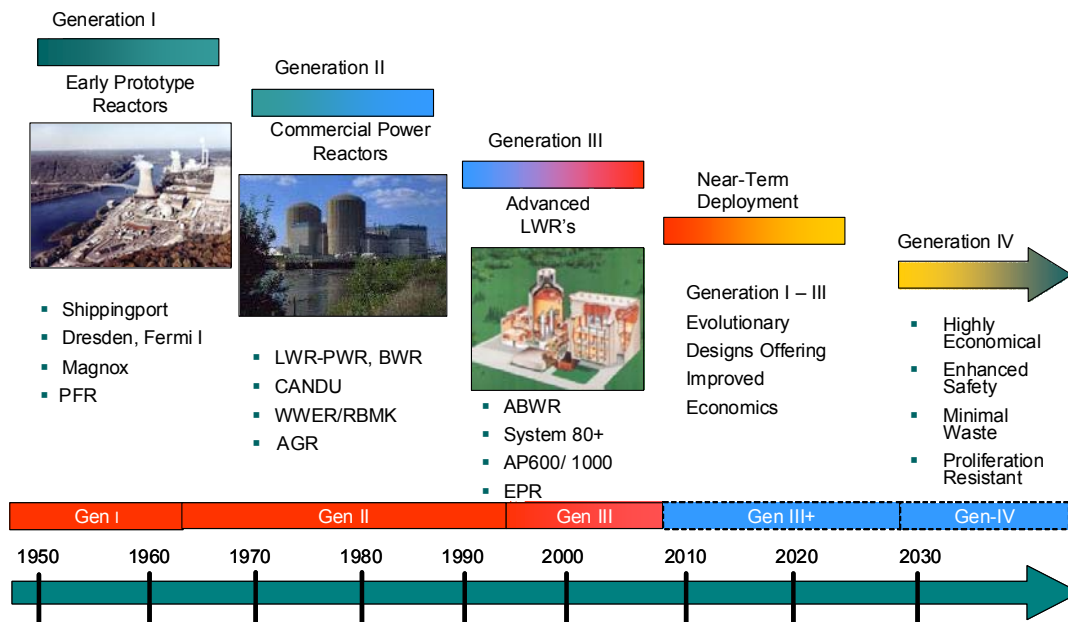


Figure 1.1 The evolution of nuclear reactor concepts

Nuclear fuels based on oxide, nitride, carbide or metal is a key issue for the development of innovative reactor systems but also for today's reactor technologies for improved economy through higher fuel burn-up. This roadmap does not however cover specific fuel materials issues since they have less of a cross-fertilization nature with the other energy technologies. There are also materials related to waste management, such as immobilization of radionuclides, but that is also not covered in this report since it is very nuclear specific.

1.1 Reactor Concepts and Structural Materials Challenges

This roadmap focuses on structural and functional materials for nuclear reactor systems. The emphasis is on Generation IV fast reactors but includes also 2nd/3rd Generation (primarily LWR) and high-temperature reactors for co-generation heat and electricity. Research needs and application areas for reactor circuit materials and components with nuclear energy specific requirements can be classified based on the stage of the reactor and the application in focus into three main areas:

- safe long term operation +60 years of existing reactors: Operation, maintenance and ageing management,
- building of new reactors, modernization of existing plants: building, construction and operational licenses
- new reactor concepts: new plant types and technical solutions, licensing requirements

In Europe, 30% of the electricity production comes from nuclear energy. Most of the operating nuclear power plants (NPP) are light water reactors (LWR) including pressurized water reactor (PWR and VVER) and boiling water reactors (BWR). Seven twin-unit advanced gas cooled reactor (AGR) are

operating in UK and four operating Magnox reactors are due to shut down by the end of 2012. Two pressurized heavy water reactor (Candu type) are operating in Romania. The 2nd / 3rd Generation LWR technology will remain the main nuclear technology for electricity generation for the next 50 years. To retain a 30% share of nuclear in the European electricity generating market in the coming decades, life-extension and power up-rate of Generation II reactors and new build of Generation III Light Water Reactor Technology will require R&D with a special emphasis on reactor materials and components. The importance of life-extension is illustrated in Figure 1.2, where the nuclear electricity production per year is shown for the European nuclear fleet with reactor shutdown after original design life of 40 years and with 20 years life extension.

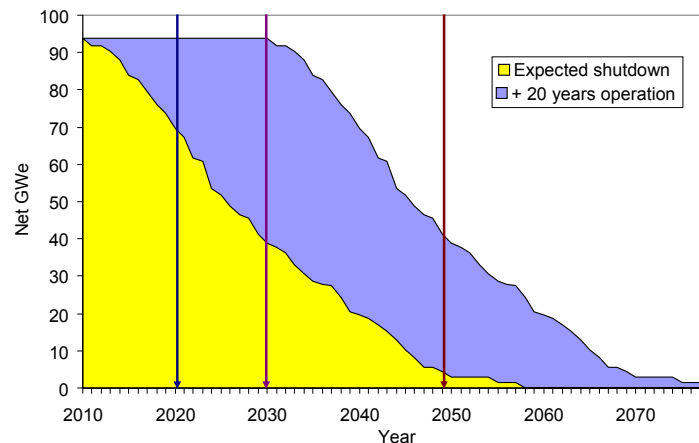


Figure 1.2 Life extension impact on nuclear power in Europe (data from IAEA-PRIS and WNA reactor databases)

The existing Generation II reactors produce reliable electricity at a low price and extension to 60 years operation with retained safety is the most important issue. Plant life management and supporting R&D activities are very important to understand and mitigate materials degradation mechanisms such as irradiation embrittlement of reactor pressure vessel steels and stress corrosion cracking of reactor pressure vessels internals and secondary components.

The first Generation III reactors are now under construction in Europe and other parts of the world. The high investment costs, high demands for development of new manufacturing methods, especially for very large components that meet the set of nuclear requirements, and the lack of knowledge and experience for such big projects has made this start rather slow.

The development of Generation III LWR reactor technologies is driven by safety (increasing passive safety features) and efficiency (e.g. size of a plant and fuel cycle), operability and maintenance (e.g. longer service cycles, modular structures, better inspection capability). All of these drivers will bring new challenges for the reactor components and materials. A key challenge is to have +60 or even +80 years design life for non-replaceable components such as the reactor pressure vessel. Another trend is to increase the fuel cycle length with increasing fuel burn up. Given the experience from the operation of Generation II and the general technology development, this opens up the opportunity to be pro-active rather than re-active to mitigate material degradation.

The recent Fukushima accident will place an even stronger emphasis on safety aspects and new safety requirements. Although, the accident was not caused by aged and degraded materials, certain material and component issues, especially behaviour under accident conditions, will need to be looked at in more detail as discussed in Chapter 4.

Nuclear energy as a long-term solution will require the design and deployment of new and more sustainable reactors, Generation IV reactors, which optimize uranium utilisation and minimise waste. These reactors need at the same time to have a very high safety level and address proliferation resistance and physical protection. They also need to be competitive with other energy systems, have high availability factor and possibly requirements for load following. To achieve these objectives, innovative design and technology features are pursued where materials play an essential role. The three key Generation IV reactor systems in Europe, and which are included in the European Sustainable Nuclear Industrial Initiative (ESNII), see Figure 1.3, are the Sodium Fast Reactor (SFR),

the Lead Fast Reactor (LFR) the Gas Fast Reactor (GFR) but also lead technology for Accelerated Driven System, (ADS)¹ for 'waste burning'. Commercial deployment is expected from 2040. In parallel there is also research for (Very) High Temperature Reactor, (V) HTR with a new generation nuclear plant (NGNP) HTR planned for construction in the US, the super-critical water reactor (SCWR) and the molten salt reactor (MSR). Table 1.1 summarizes the main characteristics of each of the different reactor types.

System	Neutron Spectrum	Coolant	Outlet temp. °C, p (MPa)	Fuel cycle	Size (MWe)
LWR	<i>thermal</i>	<i>water</i>	320 °C 7.5MPa (BWR) 16 MPa (PWR)	<i>open</i>	500-1600
SFR	<i>fast</i>	<i>sodium</i>	500-550°C -	<i>closed</i>	50-150 300 -1500 600-1500
LFR and ADS	<i>fast</i>	<i>lead</i>	480 -570°C -	<i>closed</i>	20-180, 300-1200, 600-1000
GFR	<i>fast</i>	<i>helium</i>	850°C 9 MPa	<i>closed</i>	1200
V/HTR	<i>thermal</i>	<i>helium</i>	Up to 1000°C 7-9 MPa	<i>open</i>	250 - 300
MSR (AHTR)	<i>Thermal/ fast</i>	<i>Fluoride salts</i>	700-800°C -	<i>closed</i>	300-1000
SCWR	<i>Thermal/ (fast?)</i>	<i>water</i>	510-625°C 25 MPa	<i>Open/ closed</i>	300-700 1000-1500

Table 1.1: Overview of LWR and Generation IV Systems

¹ The ADS technology is developed for the incineration of nuclear waste and in particular of Pu and the Minor Actinides Cm, Am and Np

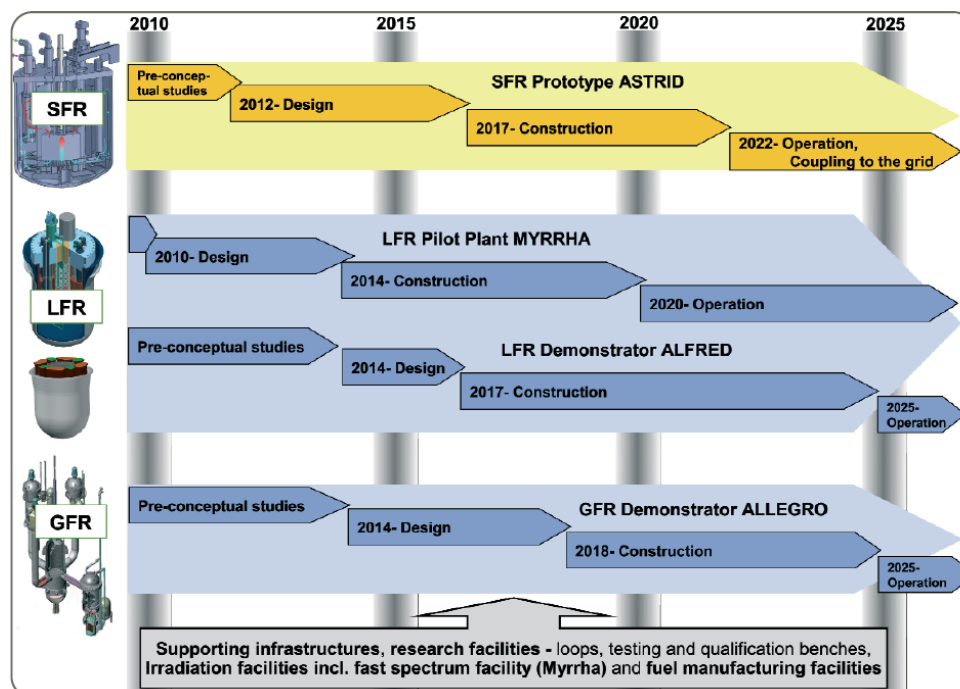


Fig. 1.3 The ESNII reactor concepts and time line for prototypes and demonstrators.

The future reactors will have higher temperatures and irradiation levels, as illustrated in Figure 1.4, and different coolants than Generation II/III reactors. This requires other materials to those used for LWRs. For the ESNII reactors the SFR is by far the technologically most developed and SFRs have been built and operated in USA, Japan, France, UK, Germany and Russia with demonstration plants up to 1200 MWe for the French Superphénix. Thus for SFR there is significant experience and the three main items for which materials development is relevant are:

1. To increase competitiveness simplified and compact design are studied. For instance, 9Cr ferritic / martensitic steels are considered for the energy conversion system to reach this objective.
2. Within the core a high fuel burn-up is envisaged, which implies the development of innovative clad materials. The reference material is the austenitic steel 15Cr-15Ni Ti stabilised. However, this material can stand burn-up levels corresponding to $\sim 100 \text{ dpa}^2$. Innovative materials as e.g. the ODS steels are here envisaged such as to reach irradiation damage of the order of $\sim 200 \text{ dpa}$.
3. Extended plant lifetime and proactive management of components and systems are two topics which are directly related to materials science and technology. In this frame, materials inspection technologies also in opaque environments need to be developed.

The lead technology (addressed for LFR and Accelerator Driven Systems, ADS) have temperature and irradiation levels comparable to the sodium technology and similar structural materials (ODS, Ferritic/Martensitic and Austenitic steels) are envisaged, but more emphasis must be put on corrosion, wear resistance and coolant compatibility issues, and protection coatings for in-core components are therefore envisaged. There is also much less experience with lead than sodium.

For GFR and VHTR, the reactor pressure vessel and the heat exchanger need to withstand high temperatures with very challenging material requirements. For the reactor pressure vessel the reference material considered is the 9Cr F/M steel and for the heat exchanger Ni alloys are under development. Finally, composites as SiC_f/SiC are reference materials for the core components of GFR and Graphite is the reference material to be investigated for the VHTR core. Table 1.2 summarizes the key components and related reference materials for the different Generation IV system. This list is not exhaustive since there are other components as e.g. pumps in LFR, fuel handling system etc.

² Dpa = displacement per atom, a quantitative measure of the irradiation a material has undergone

which would require special materials. Moreover, with respect to the reference materials listed, other materials can be envisaged and developed to address the key issues raised by the innovative design of the Generation IV systems.

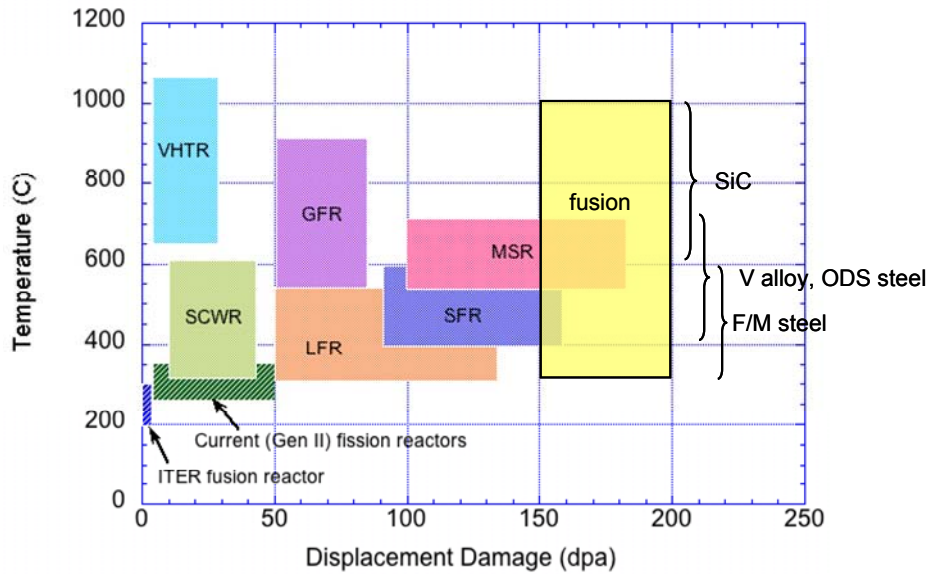


Fig. 1.4 The maximum temperature and displacement for components in different nuclear reactors.

Component	SFR	LFR	GFR	(V)HTR	SCWR*
Cladding	15Cr-15Ni Ti stab., ODS steels	15Cr-15Ni Ti stab., ODS steels	As for SFR (low power)	Graphite & Carbon Composites	chromium & nickel based austenitic steels, high nickel alloy steels & ferritic-martensitic & ODS steels
& Core assemblies	F/M & austenitic steels;	ferritic-martensitic	SiC/SiC Ceramics		
Reactor Pressure Vessel	316 L(N)	316 L(N)	21/4 Cr/ Mod 9Cr 1Mo/ 12Cr steel	SA508 or similar/ Mod 9Cr 1Mo steel	21/4 Cr/ Mod 9Cr 1Mo/ 12Cr steel
Core Support structures	316 L(N)	316 L(N)	SA508 or similar/ Mod 9Cr 1Mo steel	SA508 or similar/ Mod 9Cr 1Mo steel	316 L(N) stainless steel
Above Core Structures	316 L(N)	316 L(N)	Control Rod – as for HTR or SiC/SiC, carbon composite or Ceramics	Control Rod – Alloy 800H or Carbon composite	similar to LWR technologies
IHX	316 L(N) 9Cr F/M steel	316 L(N) 9Cr F/M steel	IN617, Haynes 230, Hastalloy X or Alloy 800H	IN617, Haynes 230 or Alloy 800H	None
Reactor Roof & Plugs	16NMD5 or similar	16NMD5 or similar	Carbon steel	Carbon steel	Similar to LWR
DRC System	316 L(N)	316 L(N)	As for HTR and VHTR	Alloy 800, ODS or carbon composite	
Steam Generator	800 Alloy 9Cr F/M steel	321 SS or similar	-???	Carbon steel or Mod 9Cr 1Mo steel	-????
Secondary pipe work	9Cr F/M Steel	16NMD5 or similar	As for HTR & VHTR	Alloy 800, ODS, or carbon composite	Similar to SC boiler technologies

Table 1.2: Key components and selected reference materials for the Generation IV systems

1.2 Challenges for process issues

The design and fabrication of structural and functional components made of different adequate materials and optimized for the specific reactor requirements will need development and application of new fabrication processes and material compositions that give enhanced material properties. Examples are thermo-mechanical treatment to produce fine dispersion of nano-sized particles in commercial ferritic/martensitic steels with superior creep properties, powder metallurgy in combination with gradient compositions and optimized consolidation procedures. As for technologies such as Spark Plasma Sintering and Electron Beam Manufacturing these too are relevant for this process and for up-scaling to larger components. Moreover for some classes of materials there is also the need to up-scale the fabrication process from the laboratory to the industrial scale. Finally, the process issues also include items such as forging, welding and joining and non-destructive investigation methods.

Returning to the three main areas above it is obvious that long service life, 60 years or more, is a common challenge for all present and future reactors concepts. This will require accelerated tests in dedicated facilities that can simulate irradiation, high temperatures and coolant compatibilities in combination with physics-based material models to interpret and extrapolate test data and basic understanding of degradation mechanisms. The long-service life requires also that in-service inspection, repair and replacement procedures are available. The building of new reactors will require that material and components are qualified according to nuclear requirements and that design rules and license procedures are available. New reactor concepts will take advantage of the experience of previous reactor development but new issues will occur. Moreover, to take full advantage of fast neutron spectra and higher temperatures requires a continuous improvement of material and component properties. The development of adequate materials for Generation IV reactors, will not be possible without a real involvement of the European industries (processing, fabrication, shaping, joining and welding of the materials) both in the materials selection and component design and importantly also the availability of nuclear facilities to test the materials in representative conditions.

2 Material Supply Status and Challenges

Primary resources, extraction rates, limits

A key resource for nuclear industry is the availability of industrialized materials technologies qualified to operate under the extreme environments of thermal, mechanical and radiation loads, in corrosive media.

The current fleet of reactors have been established at a rate of about 20 plants per year in a highly competitive industrial environment but it is now two decades since the last reactor start-up. Nuclear industries have in the meantime focused on after-sales services, ageing management programmes, refurbishment projects and life-extension from original 25 or 40 year design life to 60 years. To date well over 10.000 reactor operation years has accumulated worldwide and demonstrated safe and cost-effective longer term operation. At the same time markets in Asia have opened up, and the vast majority of current nuclear new build is taking place there, at a rate of about 10 NPP per year.

Introduction of Generation III systems is affected by a diminished industrial capacity, including loss of experience in reactor building, reduced number of players in the supply chain, as well as reduction in the volume of human resources and reduced investments. On a world scale the nuclear supply chain has become more competitive again, with new global players. The industrial basis for primary components such as pressure vessels requiring large forgings is expanding, but with the requirement for even thicker structures, this is still a near term limiting factor. Currently 200 NPP are under construction or planned, with 300 to 800 being proposed or expected to be proposed shortly.

Though uranium is relatively abundant in the Earth's crust and oceans, estimates of natural reserves are always related to the cost of mineral extraction. As the price of uranium increases on world markets, the number of economically exploitable deposits also increases. The most recent estimates identified 5.5 million tons of uranium (MtU) that could be exploited below 130\$/kg. The total amount of undiscovered resources (reasonably assured and speculative) available at an extraction cost below 130 \$/kgU is estimated at 10.5MtU. Unconventional resources (from which uranium is extracted as a by-product only, e.g. in phosphate production), lie between 7 and 22 MtU, and reserves in sea water are estimated to be 4000MtU. Japanese studies suggest that uranium from sea water can be extracted at 300€/kg. At a conservative estimate, 25000 tons of the uranium is required to produce

the fuel to generate 1000 TWh in an open fuel cycle. The global electricity supplied by nuclear is 2600 TWh, which means that the conventional resources below 130\$/kgU at the current rate of consumption would last for at least 85 years with the already identified resources (5.5 MtU) and 246 years if the undiscovered resources are also included (5.5+10.5MtU). In addition to uranium, it is also possible to use thorium, which is three times more prevalent in the Earth's crust, though this would require different reactors and fuel cycles. Nonetheless, natural resources are plentiful and do not pose an immediate limiting factor for the development of nuclear energy. However, in a scenario of a large expansion of nuclear energy, resources will become an issue much earlier, especially since new plants have at least a 60-year lifetime and utilities will need assurances when ordering new build that the uranium supply can be maintained for the full period of operation. Eventually, all known conventional reserves will be earmarked for current plants or those under construction, and this could happen by the middle of this century. This underlines the need to develop the fast neutron Generation-IV reactors.

The development perspective of the European Sustainable Nuclear Industrial Initiative (ESNII) systems and associated fuel cycles and other Gen IV systems form a clear potential and opportunity for a break-through in materials technologies. Fast reactor and high temperature systems have been built in the past, which will allow prototype Gen IV systems to be established initially with existing technologies, but full scope resources will need to be developed towards commercial deployment, e.g. through industrial initiatives. For the materials technologies needed for the advanced reactors the raw materials supply should not be critical. Availability of engineering materials on a commercial level for nuclear application is certainly a problem which became obvious e.g. for the ODS steel and for the HTR core structures. Production of high performance materials needs a critical mass for production to be interesting for a materials producer. Experimental batches and even the need for a prototype are not sufficient for a materials producer to build up or maintain a dedicated production line. This can only be achieved in concerted development/production projects (see e.g. Japanese ODS), which must be financially strongly supported at the early and intermediate stage of development.

Qualification of innovative materials and/or technologies for nuclear are costly, and will only be undertaken with solid market perspectives, and are often too large for a single demonstrator project to profit from an economy of scale. This long term perspective reaches beyond private time-horizon, and requires significant EU and national governments involvement, with tailored infrastructures. Availability of the supply chain in Europe for high alloyed steels is addressed in the chapter on fossil energy systems. Concerning fibre reinforced composites; the most interesting fibres are currently mainly produced in Japan

Secondary resources/intermediary – the nuclear fuel cycle

The spent fuel in a fast reactor system becomes a resource rather than a waste product. Through re-cycling and breeding the same amount of natural uranium can be used to produce 50-100 times more energy than today's LWR technology. Fuel reprocessing facilities have been established in France and the UK for LWRs. Markets have developed in such a way that MOx fuel has been introduced effectively for the present day LWR, and industrial facilities in Belgium, France and UK have supplied the international market. Transmutation technologies that are able to burn the waste (e.g. Am, Cm, Np) and their associated facilities are currently at the R&D stage. Fast reactors will reduce but not eliminate the amount of ultimate waste that should be finally disposed. Sweden and Finland have already selected sites for the geological disposal of spent fuel. These disposal facilities will be in operation from about 2020. France has also near-term and concrete plans for the disposal of its waste.

Supply chain

The supply of the nuclear island and the construction and engineering services which support this, are provided by specific vendors. Actual NPP construction may involve architect-engineering companies or be managed in-house by utilities. In many countries local companies are involved in the construction of NPP, mostly concerning the "balance-of-plant (BOP)" requirements, i.e. the non-nuclear specific components.

The nature of a nuclear power plant means that the owner/operator of the plant normally requires a considerable degree of "after sales" service from the vendor. In most cases, the vendor also supplies fuel fabrication services, as well as engineering and consultancy services. Services for maintenance and upgrading of existing NPPs, including major refurbishments are also provided by these vendors.

With respect to the demand from other industry sectors there is virtually no competition for the materials resources for nuclear island & fuels and BOP.

Market forces (global/EU, government/industry)

Today's new built NPP come from a range of international suppliers, with the major vendors more focused on design, engineering and project management stages. Apart from obvious economic advantages in concentrating the production of key components in a limited number of centres, there is also a notable demand from customers to maximise local supply (incl. technology transfer).

Replacement components and upgraded equipment and systems are also supplied by the vendor during the plant's lifetime. The NPP vendors are also fuel fabrication suppliers and provide most of the necessary services and components to maintain the plant through its operating lifetime. Nevertheless, the supply of fuel and other services are distinct markets from that of NPP supply.

Regulatory environments vary per country. Convergence on regulations reduces the development risk for vendors.

While many utilities do favour the original NPP vendor for these products and services, many also look to competing suppliers. All the main NPP vendors are able to supply fuel and services to plants built by other vendors, and other competing companies are also active in these markets. Nevertheless, the original plant vendor may enjoy a considerable advantage in supplying fuel and other products and services to NPPs for which it is the original supplier. Apparently this reduces the number of suppliers and vendors but it is still sufficient for a competitive market.

Material cost consideration in GenII/III plants

GenIII/II plant investment is high compared to the operation and fuel costs and more than two times the cost for other base load energy processes. The OECD NEA gives estimates for the nuclear plant construction costs with a media of 4100 USD/KWe. Risto Tarjanne gives 3400 €/kWe from a Finnish view point. The nuclear energy generation costs by NEA are evaluated to be 29 - 82 USD/MWh depending on the country. Average O&M costs are 24 % and fuel costs about 16 % out of the generating costs, refurbishment, waste treatment and 60 yrs lifetime.

Material costs are part of the investment costs in mechanical components, this is however, not openly reported. Estimates of around half are reported (2001 data).

For the large components, RPV and forgings, the major concern today is the availability of production capacity and a possible cost increase due to a peak in demand. The basic solutions are to be made with a proven technology and materials for the components that are expected to last a minimum 60 yrs in operation and are not assumed to be repaired or replaced. The associated share of investment costs is around 10 - 15 %. The material cost estimates include raw materials, manufacturing costs, inspection costs and costs for licensing. There is a need for R&D in all of these stages.

The material choices and manufacturing costs play a big role in First-of-a-Kind components but also in the maintenance phase, when replacing special components and in decommissioning of the activated components. The biggest cost effect attributed to materials may still be in unforeseen shutdowns. If a material failure due to an unexpected fast degradation or due to a failure in production/manufacture will cause a shutdown with loss of electricity production, this can have an impact on the electricity price.

An important material class is the cladding materials and materials for other components for fuel assemblies that are very critical for reactor exploitation performance. The requirements for cladding materials are very demanding and there is a challenge to develop longer fuel cycles, higher fuel burn up and better corrosion and damage resistance. The development of cladding materials would also benefit the development of new reactor concepts at the same time.

Due to the fact that irradiation is involved in the operation of all in-reactor materials the safety issues are the first criteria for all materials and manufacturing methods. For such components, the new materials and technologies will typically first be proven in other applications, where the replacement and repair are not limited by radioactive contamination. The second criteria are the operability and reliability. After passing these criteria the new material solutions should result in longer lifetime or higher operational parameters for the electricity production. Therefore, application of new materials or manufacturing methods is most appropriate in the beginning of the plant or unit life time, i.e. in new built plants or in modernization operations. Innovative tailor-made solutions will be needed for replacing components during a long life time. Also new manufacturing methods with improved surface properties or integrity can be introduced after licensing for nuclear uses.

Cost considerations for new and advanced designs (FOAK/FOAD)

Cost savings and reductions are important to the designer / constructor and the user or utility and there are important savings to be made in any system whether first of a kind (FOAK) or for follow-on advanced design (FOAD). For the design and construction phases, areas such as: reducing time to generation, manufacture and construction efficiencies, improved delivery schedules and quality control are key issues. Opportunities for improving reactor operation that are important to the user or utility include innovation that increases system effectiveness, uses non-nuclear experience to enhance availability and performance and providing improved maintenance and inspection capabilities. Some example technologies are those that simplify and quicken gaining regulatory approval, simplify and improve the robustness of the safety justification, enhance reactor lifetime and improve operational performance. Large gains can be made in areas such as maintenance (remote inspection / monitoring), non intrusive condition monitoring, reactor servicing, reduction of un-planned outages and reduction of fuel cycle costs. All these are important areas where technological development and innovation through research and development can benefit the design, enhance safety and provide important contributions in reducing costs. Research and Development too can help build up and maintain specialist reactor skills and services, services in areas such as material science, structural integrity, reactor physics, thermal hydraulics and safety assessment methodologies that will be needed to develop and support the plant over its complete lifetime (from design through to decommissioning).

For FOAK plant cost evaluations can be difficult and less precise compared with FOAD. Larger proportions of the plant construction are new and some of the materials and components will have significant elements of new manufacture and machining, supplemented by additional proving trials. In such cases material costs can often become an important part of the estimating process. For the demonstration of the Generation IV fast systems the main emphasis is on sustainability and whilst costs have to be acceptable they are not the main driver.

The levelized electricity cost from the new advanced reactors should not be higher than for Generation III or III+ plant. Although Gen IV reactors are more complex, this can be achieved since the new reactors are smaller and more modular and will in general have a more compact BOP which often represents a significant proportion of the overall Generation II/III construction costs. The size and weight of a component is important. For instance a key cost item for gas cooled systems such as the GFR and HTR will be the gas to gas heat exchanger. A conventional tube/ shell design requires a very large unit which may carry a much higher cost than a comparable steam generator. Novel, more compact, plate designs may offer cost effective alternatives. This has to be balanced against the non-repairable nature of such compact structures. Similarly increased complexity and safety requirements can also incur additional costs such as for the SFR steam generator unit (SGU) requiring the use of a secondary circuit and sodium/water reaction detection and protection devices.

Research into, and the introduction of, new and innovative designs and processes will serve to keep costs down and maintain the competitiveness of the Generation IV reactors against the Generation II and III systems in terms of the optimum balance between economics and fulfillment of their missions.

3 On-going Research and Actors

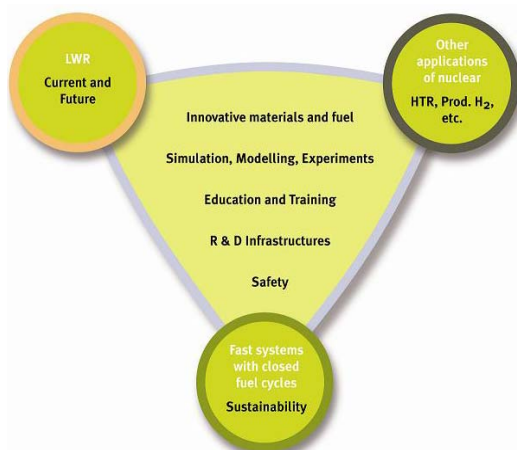


Figure 3.1 The basic three-pillar structure of SNETP

The nuclear research in the EU has traditionally mainly been performed by the individual Member States although international collaboration has also been important. The specific need for European collaboration was already identified in 1957 with the EURATOM treaty. Due to the increased cost of large research programmes, the development of a European energy policy and the globalization and increased international competition, it has become clear that Europe needs to co-ordinate its research programmes. This has led to the formation of the Sustainable Nuclear Energy Technology Platform (SNETP), which was launched in September 2007. SNETP includes industry (manufacturers and utilities) and research organizations (academia and national research laboratories), national safety organizations and the European Commission. The Strategic Research Agenda of SNETP (SRA) and the Deployment Strategy (DS) document constitute the basis for the needs and planned work in the coming decades to address the basic challenges mentioned in Chapter 1. The SNETP activities are organized into three main pillars as indicated in Figure 2.1:

1. Generation II/III light-water reactors with emphasis on life-time extension with maintained safety of Generation II reactors and design and construction of Generation III reactors with improved safety and competitiveness;
2. development of Generation IV fast reactors for commercial deployment by 2040 and
3. development and deployment of high temperature reactors for process heat applications and co-generation.

In addition there are also more cross-cutting activities relevant for all three pillars. The actual work is still mainly funded by national programmes and addresses specific needs but the formation of the SNETP has clearly led to a better co-ordination of the European nuclear fission research.

The different actors (academia, national research laboratories, utilities, vendors, material and component producers, safety authorities) have different priorities but they all have the common goal of improving safety, reducing costs and improving sustainability of nuclear energy. The different priorities and interests of these groups constitute a major challenge for knowledge management. Each Actor has an important and specific role and there are obvious dependencies. For instance the safety authority requires that the utility demonstrates sufficient safety margins against component failure and the most economical solution is to perform targeted research. In order to take into account these different priorities within a common global interest, it is important to involve all actors which is a key objective for SNETP and for most EU funded projects.

3.1 Current options being researched in EU: applied and basic research

3.1.1 Generation II/III Light water reactors

The light water reactors are based on established and proven technology plus evolutionary improvements. The two main drivers for ongoing research programmes are related to maintaining or improving safety and cost efficiency. The need for very high safety requirements and very long service life in combination with the high investment costs are key aspects. For the Generation II reactors, which were typically designed for 40 years (with the exception of Magnox type of reactors in the UK, 25years), life extension is the most important issue but improved availability and power up-rate of the NPPs are also important issues. Component refurbishment and the management of components and systems degradation are important issues that drive research and development. The evolutionary Generation III design with the larger power and hence larger components, improved safety and competitiveness require new designs and improved properties of components and materials. The Generation III reactors are designed for 60 years. Since the materials and the operational conditions are similar for Generation III and Generation II, much of the lessons learned will directly benefit Generation III. For Generation III a more pro-active approach can therefore be taken to address degradation mechanisms at the design stage. An overview of materials and key degradation mechanisms is given in Table 3.1.

Component types	BWR material examples	PWR material examples	Degradation examples
Reactor Vessel	Bainitic steel with austenitic cladding 309SS	Bainitic steel with austenitic 308, 309 SS cladding	Irradiation embrittlement
RPV internals	austenitic stainless steels (wrought or cast SS). Ni-based alloys (750X)	austenitic stainless steels Ni-based alloys	Irradiation embrittlement Fatigue, Environmentally Assisted Cracking (EAC, IASCC)
Steam generator	not valid	ferritic fine grained steel steam dryers 304SS, tubing 600MA, 600TT, 690TT, 800	EAC
Steam and Water piping, vessels, valves	cast duplex SS, ferritic steel SS cladding, ferritic steel	cast duplex SS, ferritic steel SS cladding, ferritic steel	Fatigue, EAC
Other	Condenser: carbon steel, tubing Ti, SS and pre-heater SS	Condenser: carbon steel, tubing Ti, SS and pre-heater SS	Fatigue, corrosion, EAC
Fuel cladding Control rods Fuel assembly	Zr-2 304, 316 SS, B4C	Zr-4, advanced Zr alloys Ag/In/Cd control rods SS Clad, B4C+SS Inconel 718	EAC, IASCC, fuel pellet interaction, creep

Table 3.1 Overview of materials and degradation mechanisms for Gen II LRW

The Network of Excellence, NULIFE, has been very important in recent years for focussed research on materials issues for Long-Term Operation (LTO) of light water reactors. The broader R&D areas dealt within the NULIFE and FP7 umbrella projects address:

- European harmonised plant design and safety justification methodology,
- Integrity assessment of the main components,
- Ageing mechanisms of Structures-Systems-Components to build up predictive models
- Ageing monitoring to allow proactive degradation management,
- Prevention and mitigation of ageing by applying the best practice,
- Pre-normative research, codes and standards.

The on-going projects are to a large extent focussed on understanding material degradation mechanisms to reduce the uncertainties related to LTO. Materials issues must be resolved for reactor pressure vessel, core internal, primary piping, weldments, concrete, secondary systems, cables and buried pipes. Key issues that are under study:

- Prediction of radiation embrittlement and its impact on reactor pressure vessels integrity has been and remains the most important research issue. It has very important safety implications as the vessel is the key safety barrier. It has also economical implications since the RPV is irreplaceable and decides the life-time of the reactor. The research include optimizing surveillance programmes, development and application of adequate models such empirically based model, e.g. Master curve for ductile-to-brittle transition as well as advanced physics-based models to simulate irradiation damage.
- Environmental assisted and irradiation assisted stress corrosion cracking for in-core austenitic steel components. The objective is to increase the knowledge and develop prediction models for long term operation, and during transients, similar to the case of RPV steel embrittlement.
- Environmental effects on mechanical properties. This includes environmental effects on fatigue code curves for stainless steels and low temperature crack propagation of nickel-based weld metals.
- Stress corrosion cracking in austenitic steels and Ni-based alloys. This problem is very complex since it depends on material, loads and the water chemistry. Large experimental programmes that cover parameters are therefore very important but there is also improvement in more fundamental models.
- Fuel assembly materials performance under normal and accidental conditions. Zr alloys for fuel cladding but also Ag/In/Cd alloys, B₄C, austenitic steel, Ni alloys for internal components. Important issues include in-pile deformation, irradiation induced swelling, degradation of cladding properties from higher burn-up, pellet-cladding interaction.
- Prediction of weld integrity and in particular dissimilar metal welds. Welds often constitute the weakest areas in nuclear components. Although the trend is to reduce the number of welds in newer designs they cannot be avoided. Major issues include understanding how weld procedures affect mechanical properties of the welds, stress distributions and defect formations.
- Thermal fatigue due to stratification and turbulent flow in pipe connections. Research includes treatment of complex load spectra, fluid-structure interaction and complex crack configurations.
- Ageing of non-metallic materials such as concrete, buried piping and cable insulation
- Monitoring, replacement and repair of degraded components.

NULIFE has launched several projects funded by partners and sometimes with EURATOM support. Recent and ongoing projects include:

- Irradiation embrittlement, highly advanced physics based models to assess irradiation effects (PERFECT and PERFORM 60); treatment of long-term embrittlement effects in RPV (LONGLIFE),
- Improved structural integrity and safety margins for metallic components. probabilistic approach for lead-before break (LBB); constraint and biaxial loading effects including warm pre-stress (CABINET and NESC VII); development of structural integrity assessment for reactor coolant components (STYLE); Dissimilar metal weld integrity (NULIFE DMW), European procedures for thermal fatigue (NULIFE and NESC Thermal fatigue).
- Stress corrosion cracking (NULIFE pilot on SCC).
- Ageing of concrete and civil structures (ACCEPT).

3.1.2 Generation IV Fast reactors

The European context is even more important for the development of future reactors and the establishment of the European Sustainable Nuclear Industrial Initiative (ESNII) is a response to this need (Figure 1.3). European research in the design of the next generation of fast reactor systems is performed in CP-ESFR, LEADER and GoFastR for SFR, LFR and GFR reactors respectively. In these projects the basic designs are proposed including materials for specific components. The demonstrators and ESNII prototypes for SFR, LFR and GFR and the supporting infrastructures will be crucial to demonstrate the viability of these reactor concepts. The most relevant European initiative that specifically addresses materials for nuclear energy is the Joint Programme for Nuclear Materials in the frame of the European Energy Research Alliance (EERA), which was launched in November 2010. This programme gathers 15 research associations and universities from 11 European countries. An effort of potentially 135 Person*year/year over the next five years has been proposed by the partners to address key items identified. The overall objective is to validate, screen and develop/improve candidate materials for the ESNII reactor systems. This is done within four Sub-programs:

1. support to design and construction of the ESNII demonstrators with emphasis on commercially available materials (austenitic steels, ferritic/martensitic steels, nickel-based alloys);
2. ODS steels development;
3. development of refractory materials, ceramic composites and metal-based alloys;
4. modelling: correlation, simulation and experimental validation.

Two further important ongoing projects in the European frame addressing materials for Generation IV systems are the GETMAT (Generation IV and Transmutation Materials) and the MATTER (Materials Testing and Rules) EC supported projects. The GETMAT project has its main focus on ODS steel characterization and development and experimental validation of Fe-Cr physical models. The five years GETMAT project started in 2008 and foresees an effort of about 20 Person*year/year. The MATTER project focuses on pre-normative research of 9Cr F/M and austenitic steels for test procedures and Design Rules to identify properties and gaps needed for the nuclear standardization. Moreover, in the MATTER project a small activity on ODS steel is included, where the fabrication process is investigated with the aim to improve fabrication procedures and materials quality. The four years MATTER project started in 2011 and foresees an effort of about 22 Person*year/year. Both GETMAT and MATTER projects represent a collaborative approach between research associations and industry. The development of a European Design Code based on the French RCC-MRx is also being undertaken under the auspice of CEN. Test facilities for material characterization and qualification are indispensable for the development of future nuclear reactors. The FP-7 Co-ordination and Support Action ADRIANA, addresses this by a detailed roadmap, including legal and financial solutions for the research infrastructures needed to develop next generation fast neutron systems such as hot cells, fast-flux irradiation facilities, loops to study material compatibility with the different coolants. The roadmap addresses major refurbishments, upgrades of existing and construction of new facilities, trans-national access and related education and training issues. F-Bridge, is also an important EU project that addresses basic research for fuel and the claddings for the next Generation IV reactors and with special emphasis given to transfer between more basic research and technological applications. Most of the research for future nuclear reactors is still performed within the different national programmes, with the most comprehensive programme in France.

3.1.3 High Temperature Reactors

The R&D on temperature reactors and con-generation in Europe has mainly been initiated through the European High Temperature Reactor Technology Network. Recently the efforts have concentrated on high temperature reactors (HTR) with process heat up to 850 C, rather than the very-high temperature reactors, (VHTR) with process heat of 1000 C or more, mainly due to the materials related problems.

For materials issues and on the European scale the **HTR** development has been promoted by the **RAPHAEL** (ReActor for Process heat, Hydrogen And ELectricity generation) project and a new

project ARCHER. Near term challenges include selection of graphite for the core, development of intermediate heat exchanger (IHx) and the development of a process heat coupling system (investigated in EUROPAIRS) since it needs to be coupled to a non-nuclear industrial process. The qualification of candidate materials, graphite, nickel based steels etc, are very important. The co-generation community within SNETP is presently preparing a European Industrial Initiative for a research programme aiming at demonstrating the feasibility of co-generation for intermediate temperature. A major issue here is the coupling between the nuclear heat generation and the non-nuclear end user of the heat.

3.2 Comparison with on-going efforts in other developed countries

There is international collaboration on the development of nuclear technologies through international organizations such as IAEA and OECD/NEA. For Generation IV, there is also an established international collaboration through Generation IV Forum, GIF. Close interaction collaboration is vital to deal with the nuclear issues as one accident in one region affects the global support (Harrisburg, Tchernobyl and Fukushima). An example of international collaboration is that utilities from USA, Japan, and France, operating more than the half of the world's fleet have joined their effort by launching the "Material Ageing Institute" in France, which has as a main objective to conduct research and development related to ageing of materials used in present nuclear power plants, as well as training and education of engineers and students in this field.

The most important regions outside Europe with significant nuclear materials research are North America (USA and Canada), Asia (China, India, Japan and South Korea) and Russia.

The life extension of both BWR and PWR reactors is the most important nuclear activity in the US. This has triggered life extension research similar to the Generation II research in Europe. The USA does not have a large national demonstrator programme for fast reactors but US organizations and scientists are still active contributors to the basic materials research through US Department of Energy (DoE) funded research, with an emphasis on modeling. The US is actively pursuing a project to develop a high-temperature reactor for heat process and co-generation (NGNP).

Most reactors that are under construction or planned are in China. China has very ambitious plans to develop its scientific and industrial basis in nuclear energy. Light water reactors of different designs, including European, US, Japanese and Russian designs are under construction. These projects are often joint-ventures and a large proportion of components are manufactured in China. Gradually China is developing its own nuclear industry. China has also research reactors for HTR and sodium fast reactors. Clearly China will be a very strong future nuclear operator and competitor.

India has a flourishing and largely indigenous nuclear power program and expects to have 20 GWe nuclear capacity on line by 2020 and 63GW MWe by 2032. It aims to supply 25% of electricity from nuclear power by 2050. India has a vision of becoming a world leader in nuclear technology based on its expertise in fast reactors and thorium fuel cycle. India already has an active programme for building SFR fast reactors of a technology similar to the European EFR programme in the eighties (hence they are not in Generation IV) and materials similar to the SFR demonstrator. It is likely that this will give India a leading role in the development of fast reactors. India is building up its competence in both basic nuclear research and industrial capacity.

Japan is traditionally a leading nuclear nation and has a large market share for Gen II and III reactors (both PWR and ABW). The Japanese industry is leading the development of materials and components. For instance it is presently the only nation that can produce very large forgings for pressure vessel heads. Japan developed fast reactor technologies from the seventies and is now building a fast reactor demonstrator of Generation IV, with timing in parallel with the ESNII. The accident in Fukushima will have a large impact on the Japanese nuclear programme and the large expansion (from 35 to 50% of the electricity) that was planned will probably be reduced.

Russia is steadily moving forward in the nuclear area and has a clear ambition to become the world leading nation. Russia has an ongoing programme for life-extension and power upgrade of its Light water reactors and more plants are presently being built or planned. Russia has a strong position in fast reactor technology for both SFR and LFR. In LFR technology they are probably the world's leading nation, largely because of their reactors for nuclear propulsion. A modular lead-bismuth

reactor is also planned. Russia is the only country who has operated fast reactors on a commercial basis (BN-600 MW). Russia plans a gradual transition to fast reactor technology and a BN-800 reactor will replace the BN-600 in a few years and then this will be followed by a BN-1200. The large nuclear programme also means that Russia has a strong position in the nuclear materials research.

4 Materials Specification Targets for Market Implementation in 2020/30 and 2050

4.1 Reactor Types and deployment

The different types of reactor will need to cope with a range of operating conditions (see Table 1.1 and Figure 1.4) and load cycles and require materials capable of withstanding the different environmental conditions they experience. The candidate materials for key components of the different reactor concepts are summarized in Table 1.2 for Generation IV and Table 3.1 for Generation II/III.

4.1.1 Generation II, III and III+ Systems

These systems include the existing water cooled and gas cooled systems and the new near term Generation III+ reactor constructions. Materials requirements and developments are expected to build on past experience and to a large extent depend on the knowledge gained from more recently constructed reactors such as Sizewell B, the EPR and AP660/1000 plant constructions.

These reactors are based on a mature technology with main challenges in control and safety assessments rather than in materials. Increasing burn-up and improving efficiency might necessitate advanced cladding or new routes for materials processing which might not be fully achievable in the shorter time-frame. However a key component important to safety and requiring continuing R & D is the reactor vessel. The primary containment barrier has to be maintained for all loadings and over the entire lifetime. A major life limiter is the degradation mechanisms that can cause cracking and reduction in strength and it is important to continue to address these in the near term and especially as the design life is extended.

In the aftermath of the Fukushima accident there will be even stricter requirements in the coming years to demonstrate the integrity of fuel claddings and the reactor vessel under severe accident scenarios.

The Generation III reactors need to be designed for 60 years and should address the major degradation mechanisms at the design stage. The larger components will also require new procedures and solutions.

4.1.2 The ESNII Fast Reactor Concepts: SFR, LFR and GFR

Structural materials for the three fast reactors concepts in ESNII will be exposed to higher irradiation damage and temperatures than light-water reactors and operate in corrosive reactor specific coolants and with a design life of at least 60 years for non-replaceable components. The timeline for the deployment is shown in Figure 1.3. The prototypes and demonstrators are expected to be in operation by 2020/30 and commercial deployment from 2040. The SFR prototype ASTRID and the eutectic lead-bismuth reactor, MYRRHA, which serves as a pilot technology plant for LFR, provider of fast neutrons for materials qualification and a demonstrator for the ADS technology, are at an advanced planning stage. The operating temperature and neutron displacement damage regimes are shown in Figure 1.4. The fuel cladding is the component that will experience the highest operating temperature and irradiation levels (e.g. 600°C and 150 dpa for SFR) whereas for the vessel it will be lower around 400°C and exposed to less than 1 dpa. The components will also need to retain their integrity under postulated reactivity induced accidents with high temperature transients.

The main material requirements for these future reactors are:

- In-core materials need to exhibit dimensional stability under irradiation (irradiation creep and swelling) and high temperatures.
- Basic mechanical properties (tensile strength, ductility, fracture toughness, creep and fatigue resistance) need to remain acceptable during the entire design life.

- The properties need to be retained under the corrosive environment specific for the different reactor types.

Some specific key issues that need to be addressed in the near future include:

- Creep-fatigue for out-of-core components at higher temperatures. Creep-fatigue of some materials is not well understood and there are no accurate assessment methods. This applies to 316 austenitic steels and to a greater extent to ferritic-martensitic steels that experience both cyclic softening and significant creep.
- Material properties such as creep rupture, can be significantly inferior at welds compared with base material. Basic physics-based models, design and assessment criteria as well as welding procedures need to be developed for the range of applicable welds and conditions.
- Both austenitic and ferritic-martensitic steels show significant loss of hardening and ductility when irradiated to small neutron displacement levels at low temperatures ($< 350^{\circ}\text{C}$) which will be important for components such as vessels.
- Advanced protective coatings are needed to reduce wear and corrosion for key steel components and extend their operational range and lifetime
- Compatibility between the component and surrounding medium, for example:
 - Liquid-metal embrittlement, wear and erosion, stress corrosion and corrosion fatigue in liquid lead and in particular in combination with irradiation (T91 and 15-15Ti) (LFR)
 - Water/steam oxidation, long-term corrosion in Na, hydrogen embrittlement at elevated temperatures (SFR)
 - Incompatibility with He impurities at high temperatures for in-core and out-of-core components (GFR).

More detailed descriptions of these systems can be found in the Appendix.

Nuclear energy provides primarily base load electricity. In a future low-carbon energy mix with a large share of intermittent renewable energy sources (e.g. wind and solar), it will be necessary that nuclear develops some load following capacity. This is technically feasible today, and also applied in France to a limited extent. In ESNII load following has therefore been identified as key performance indicator with a load following capacity of 10% and 20% per minute of the full power as “realistic” and “optimistic” targets. Load following would mean that a larger number of load cycles need to be taken into account in the design of the components.

4.1.3 The other Generation IV Concepts: (V)HTR, SCWR and MSR

HTR

The HTR system is helium cooled and graphite moderated and uses a thermal spectrum reactor with an elevated core outlet temperature. The reference thermal reactor power allows passive decay heat removal and the capability for the co-generation of electricity and hydrogen enabling the reactor to be used for process heat applications. The reactor system will use materials such as ferritic/ austenitic steels and nickel alloys (such as Alloy 800, IN 617) for the main components. The graphite core will require one or more new graphite materials to be fully qualified since the graphites used for the cores in the previous gas cooled reactors (AVR, AGR, etc.) are no longer available. Except for graphite, materials developed for the HTR system have largely spearheaded the materials that are to be used in the GFR. For the vessel in the near term, HTR is more likely to make use of a cooled PWR type vessel than a non-cooled ferritic-martensitic steel (Mod 9Cr 1% Mo) vessel which has been investigated for VHTR. The GFR system on the other hand will still require a ferritic-martensitic steel development which will still need to be investigated for 850°C temperatures. There are plans to build a prototype for HTR for co-generation in parallel with the ESNII fast reactors.

SCWR

The SCWR has coolant temperatures in the region $520/ 580^{\circ}\text{C}$ and operates at much higher supercritical pressures than the Generation II/III systems. For the SCWR main challenges include cladding materials, the development of special testing facilities to study the nature of SCW and effects of radiolysis for design, and the development of a suitable chemistry control strategy. The material selections are similar to those of novel BWRs and PWRs and irradiated assisted stress corrosion cracking (IASCC) and stress corrosion cracking (SCC) under SCW conditions are a major degradation mechanism. Although some results have been reported, more sophisticated methods for

loading and monitoring the autoclave tests are needed at temperatures over 550°C in SCWR environments to simulate cladding conditions. The higher neutron absorption makes Ni-based alloys less favorable than stainless steels for internal components due to IASCC. Identification of appropriate SCWR chemistry and materials for in-core and out-core components has been carried out over the last 10 years. There is however still insufficient data available for any single alloy to ensure its performance under the SCWR conditions. Results from SCWR investigations on IASCC and SCC will help to support extrapolations of conditions for austenitic and Ni based alloys in Generation III+ reactors and the development of methods for loading and monitoring in autoclave tests. SCWR is sometimes considered as Generation III+ (SNETP) rather than Generation IV (GIF), and it would be technically feasible to build a prototype in the coming decades.

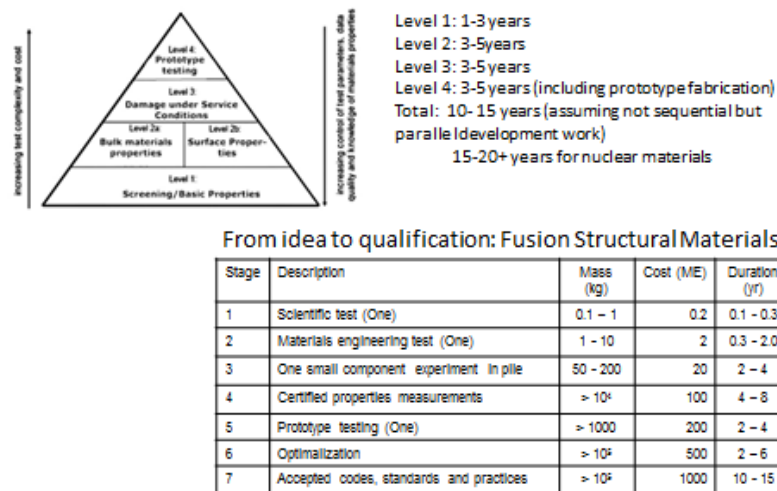
MSR

The Molten Salt Reactor (MSR) uses a fuel dissolved in fluoride salt coolant with the ‘fast reactor version MSFR recognized as a long term alternative to solid fuel systems. Molten salts also provide opportunities for intermediate heat transport in other systems (SFR, LFR, VHTR) giving advantages in using smaller sized equipments (because of higher heat capacity of the salt) and the absence of chemical reactions between the reactor, intermediate loop and the power cycle coolants. Liquid salt chemistry plays a major role in the viability of the MSR and MSFR with important R & D issues still to be addressed on the physical and chemical behaviour of the coolant and fuel salts and the effects of fission products and of tritium. Other issues include the compatibility of the salt with the structural materials and fuel processing material development, and the general issues of maintenance, instrumentation, chemistry control and safety aspects. There are no plans to build a MSR demonstrator in the near future.

4.2 Material deployment

4.2.1 Basic requirements

Materials Design and Implementation



From idea to qualification: Fusion Structural Materials

Stage	Description	Mass (kg)	Cost (ME)	Duration (yr)
1	Scientific test (One)	0.1 – 1	0.2	0.1 – 0.3
2	Materials engineering test (One)	1 – 10	2	0.3 – 2.0
3	One small component experiment in pile	50 – 200	20	2 – 4
4	Certified properties measurements	> 10 ⁴	100	4 – 8
5	Prototype testing (One)	> 1000	200	2 – 4
6	Optimization	> 10 ⁴	500	2 – 6
7	Accepted codes, standards and practices	> 10 ⁴	1000	10 – 15

Figure 4.1: Steps for design and implementation of structural materials for advanced (nuclear) applications

The process of development/selection of structural materials for plants and components acting under extreme environments involves the following basic requirements:

- Materials must be suitable and commercially available,
- Component design and manufacture must be possible at affordable cost,
- Trusted suppliers must exist,
- Repair or replacement must be possible,
- Design must be covered by design codes,

- It must be demonstrated that components retain their function and integrity during their design life,
- Non-destructive monitoring must be possible,
- Degradation mechanism and failure probabilities must be assessed,
- The solution must be accepted by authorities and customer.

This wide range of requirements need considerable development times as indicated below in Figure 4.1. The table shows data from fusion experience concerning a modification of a martensitic 9-12% Cr steel.

Taking delays in the nuclear acceptance including eventual supporting test and development work into consideration the 20 years indicated could be optimistic and extend to 30 and even more years of development. As a result it follows that for market implementation 2020/30 no completely novel materials are expected, but improvements of currently existing (and not yet introduced) materials as well as safety and design related issues need urgently to be resolved. It should also be realized that because of this lengthy implementation period development work for materials for implementation in 2050 has to start in the 2020/2030 time frame otherwise these materials will not be ready in time.

4.2.2 Materials for nuclear components

Austenitic steels

Austenitic steels, such as 316L(N), have good creep properties and corrosion properties at moderate temperatures. They are widely used in today's reactors and there exists extensive operational experience. They are also important for Generation II and III systems. For the fast neutron systems austenitic steels will be used in both the reactor block and primary circuit. A class of austenitic steels, 15-15 Ti, is also used for the cladding within the core. Further development includes alloy improvements and ensuring a better performance under irradiation conditions. With regard to strength, increased long term creep strength for extended life times for base material and welds will be important.

Ferritic/martensitic steels (9-12% Cr)

These materials will be widely used in Generation II/III and Generation IV reactors. In ESNII, Modified 9Cr-1Mo steels are the reference material for the different key components because of their good resistance to corrosion and wear, good creep properties, low irradiation swelling, high thermal conductivity and low thermal expansion. They are commercially available today although there is less experience than with the austenitic steels. They should be developed further to provide improvements to properties to enable operation at higher temperatures. As shown by past experience such an approach represents a cheaper option to moving up to the next range of materials. An example of where such benefits have occurred (in conventional plants) is the development of the 650°C ferritic steam generator. In the longer term developing materials to withstand higher temperatures (e.g. 700°C or higher) could provide a significant cost benefit to industry and plant operators. It has been demonstrated that significant improvement of the creep properties can be achieved by thermo-mechanical treatment processes. The need to construct long heating tube lengths also necessitates the use of alloys with good manufacturability and weldability and the more highly alloyed ferritic steels are the leading candidates for this role.

Ni-based superalloys

Nickel-based super-alloys (Haynes 230, Alloy 617, Alloy 800H) are generally used for heat exchangers and steam generators at outlet temperatures <750°C with creep strength requirements of the order of 100 MPa after 10⁵ h conditions. At higher temperatures the long term creep strength is insufficient to provide reliable service given the nuclear fission requirement of few numbers of maintenance periods and continuous operation at full temperature for periods of 20 years or more. They have rather poor irradiation resistance which restricts the use to low irradiation components. There has been little demand over the last few decades for development of higher temperature versions of these materials and much is still required with regard to their acceptance and application

to large-scale component manufacture. It is also unlikely that a nuclear renaissance of advanced nuclear systems will, by itself, be sufficient to fuel demand.

ODS steels

Oxide Dispersion Strength (ODS) steels give increased creep strength at high temperature compared with Ni based alloys (e.g. Alloy 617) and the 9Cr ODS steel is considered to be a promising option for the SFR fuel cladding material particularly for high burn-up and higher temperature operation. ODS steels suffer from a low fracture toughness at relatively low temperatures. Currently there is no European supplier of ODS materials for nuclear application and development or research and manufacturing and joining issues remain a key challenge to their use. Establishing ODS materials on an industrial footing will require a significant investment and a close co-operation with research, industrial users and manufacturers. It is unlikely that such materials will find a nuclear application in the 2020/2030 term. Nevertheless the research into these materials needs to start to establish them on a stronger footing for potential application in higher temperature more advanced versions of the Generation IV systems for 2050 and beyond.

Ceramics/composites (Graphite, SiC/SiC)

Current envisaged applications range from the reactor core to the heat transferring pipes and heat exchangers. Graphite is selected for the core of the helium cooled HTR. Investigations undertaken within the FP6 and FP7 European Framework programs represent the state of the art within GIF on graphite properties and these results provide a basis for selection from currently available graphites. However, should none of the current graphites undergoing qualification turn out to be suitable candidates for future HTRs, a programme of graphite development/qualification will be required in this time frame.

Carbon composite and carbon fibre materials and SiC/SiC materials can provide increased resistance to temperature and irradiation in comparison with available metals and are therefore potential candidates in the longer term for the reactor core components and the control rod. A drawback is the low fracture toughness. These materials are available today for non-nuclear applications but can be expensive to manufacture. More importantly nuclear application regulatory acceptance is required and screening and selection programs involving irradiation testing, corrosion measurements and architectural optimisation (for composites) and modeling are necessary to obtain the best economically viable material for the required application.

Coatings

Protective coatings or other surface modifications can provide increased resistance against corrosion and wear and thermal shocks and hence increase the design life and reliability of nuclear components subjected to severe environment. The coatings must withstand repeated thermal cycling under long-term exposures to high temperature, chemical aggressive environment and irradiation. Coatings are therefore essential for all reactor concepts and increasingly for future reactors and for different components and materials. The performance of the coating system depends of course on the coating material itself but also on the interface with the substrate. Stellite, a cobalt alloy coating, has been used for LWR. But because it is activated by neutron irradiation, it is not favoured for fast reactors and the ASTRID project is now investigating alternative coatings. Coatings are also tested for P91 fuel claddings and FeAlCrY coating is a reference for corrosion protection. Coatings are also important for composites and graphites and SiC coatings can improve oxidation resistance to graphite for high temperature reactors.

Other materials (e.g. intermetallics)

This category covers other possible materials including intermetallics, concrete, and other cast materials. Titanium Aluminides were developed as high temperature structural materials mainly for automotive and gas-turbine applications. The first main development fell into a period of decreasing interest and so they were never seriously considered for nuclear applications, except for a few investigations into the irradiation behaviour. Extended characterization of α_2/γ -TiAl has shown that these materials have a creep rupture strength (in vacuum) exceeding that of IN617 by a factor of two (in stress). The irradiation creep performance was found to be comparable with martensitic and ODS materials. However the main obstacles for their introduction are low toughness at room temperature and a tendency for oxidation in air combustion gases at temperatures above about 800°C. Like

superalloys this class of materials tends to embrittle in helium at high temperatures which is mainly due to helium bubbles. Introduction of these materials in the 2020/2030 time frame is unlikely for the same reasons as for ODS materials. Nevertheless they should be re-evaluated in the near term to confirm whether they represent viable alternatives for the 2030/ 2050 higher powered Generation IV reactors.

Concrete is widely used in Gen II and Gen III systems as a containment (boundary) material (with reinforcements) and as a structural material for the buildings and for the (seismic) support rafts. Specialist concrete materials have also been developed to resist temperature for application in high temperature fast and thermal systems. Whilst much is known and has been developed in terms of understanding of behaviour and modeling there is still a lack of suitable material models for high temperature and work needs to be done in this area for application to the Gen IV systems. In addition to concrete other cast materials such as cast iron again reinforced with ductile metals can also provide an alternative solution for structural containment. These materials are brittle but can resist higher temperatures. Such materials may see increased deployment in the 2020/2030 time frame where they offer a viable alternative to other more expensive solutions.

4.3 Targets for 2020/30 and 2050 Deployment

Given the number of candidate materials and the reactor specific conditions the most urgent overall need is to develop methods and a database for material screening and qualification to assure integrity and functioning for the 60 year design life. This will require additional data on the combined effect of high temperature, high irradiation levels and coolant compatibility. A key generic issue is also to establish test methodologies for accelerated tests and small specimen tests and for extrapolation to service conditions. This will need seamless integration of physics-based models and experiments in dedicated facilities. Equally important is to develop robust but not overly conservative Standards and Design Rules and their codification. The development of on-site monitoring and damage prevention systems are also important aspects. For materials that are commercially available there will be a need to improve their properties by alloy and microstructure modifications. For instance the maximum operating temperature for steels has increased by 2.5 °C/year for the last 60 years and the strength of pressure vessel steels has increased by 50%. The less developed materials need to be developed at laboratory scale and subsequently upscaled to industrial scale. Issues associated with irradiation, long-term creep, ageing and corrosion will also need to be examined including long term weld performance and for extended service feedback of relevant and reliable material properties from past operation and testing.

4.3.1 2020/30 Targets for Market Implementation

Global Targets

- Addressing material issues directly related to the Fukushima accident. This could include primary corrosion and chemistry and steam/air oxidation of zirconium alloys and development of fuel claddings that do not produce hydrogen through e.g. coatings) and in longer term non-metallic claddings (e.g. SiC/SiC)
- Qualification of the materials needed for the construction and licensing of the ESNII demonstrators. This is primarily addressing the commercially available materials (austenitic steels, ferritic-martensitic steels and nickel-based superalloys), but includes also pilot versions of ODS steels for fuel claddings.
- Development of European Design Codes for the advanced reactor systems. This should be based on an extension of existing Design Codes (e.g. RCC-MRx, R5, R6, ASME).
- Qualification of graphite as core material for helium cooled HTR at intermediate temperatures (up to 800° C)
- Basic screening of the behaviour and performance of ODS steels under the relevant conditions and development of basic design curves. Development of fabrication and welding of ODS steel components going from laboratory scale to a pilot scale.

- Screening and selection program for ceramics/composites and SiC/SiC involving irradiation testing, corrosion measurement and architectural optimization. A set of design curves giving properties under the application operating conditions will need to be developed.
- Re-evaluation of other materials (e.g. Titanium Aluminides) to confirm whether they represent viable alternatives for the 2030/50 higher powered Generation IV reactors.

Specific Targets

- Development of a methodology to understand long-term degradation through accelerated tests. This applies to all materials.
- Development and standardization of miniaturized test procedures for semi-non-destructive testing, weld materials, irradiated materials
- Improvement of fracture toughness of ODS steels and ceramics and composites
- On-site monitoring and damage control and prevention programme for Gen II and III nuclear reactors.
- Basic understanding of fuel-cladding chemical and mechanical interaction and associated safety related degradation mechanisms.
- Improvement of material properties that limit the exploitation and deployment of specific nuclear reactor systems. These specific property requirements would result from the screening and selection programmes. This may include thermo-mechanical treatments, alloying, coatings to improve heat, corrosion and irradiation resistance.

4.3.2 2050 Target for Market Implementation

Global Targets

- Qualification of the materials needed for the construction and licensing for the deployment of commercial innovative reactors. This involves all the relevant materials (austenitic and ferritic-martensitic steels, nickel-based alloys, ODS steels and composites and ceramics).
- Up-scaling of the fabrication and welding of ODS steels to a full industrial scale. A particular need is to develop monitoring techniques for mechanical alloying to determine the main parameters to be controlled during the up-scaling process and after the transformation step, to develop non destructive methods at the scale of the potential defects. Joining technologies that avoid melting are needed such as resistive joining or Friction Steer Welding. Even if mechanical alloying becomes the basis for future ODS steel development it is important to investigate more exploratory high potential processes such as EB-PVD (Electron Beam Physical Vapor Deposition) which is a coating technology, but can also be used for vapor forming three dimensional parts. This technology has for example been developed by the Harbin Institute of Technology (China) to produce nickel based ODS foils and it has the added advantage of being easily up-scaled (as has been the case in the aeronautics field), and offers the potential for the development of gradient materials with surface properties that are different from the bulk.
- Up-scaling of nuclear grade composite materials at an industrial scale. This includes the need to develop gas tight materials to resist gaseous fission products although there is a possibility of integrating a liner using for example the CVI (Chemical Vapour Infiltration) process which could constitute a first solution. The problem of chemical reactivity between the SiC/SiC and the fuel also has to be solved as does the large fabrication costs of nuclear grade CMC. The reference process technology is CVI, but nanotechnology could also provide a promising alternative production route at lower cost (e.g. use of SiC nanopowders slurries in order to infiltrate SiC textures). The synthesis of the nanopowders can be carried out either by plasma technology or by laser assisted technology and the development of a nanoparticle based process will require parallel developments in associated tools for a safe nano-manufacture. The process development also needs to avoid the grain growth during sintering. Spark Plasma Sintering (SPS) process offers a promising technology in this respect. SiC/SiC development at a European level also needs to

consider strategic issues such as the production of high performance SiC fibers which up to now have been exclusively developed in Japan and whether to promote a European industrial chain to solve this potential problem.

- Qualification of graphite for VHTR with $T > 950^{\circ}\text{C}$.
- More advanced versions of the materials that are commercially available already could offer a viable alternative to more expensive materials. This objective would be to produce versions with successfully higher temperature and irradiation resistance.
- Further advancement in Design Codes to address materials and conditions for higher power and higher temperature Generation IV reactors and VHTR.
- On-site monitoring and damage control and prevention programme for innovative reactors.
- Development of advanced predictive models based on fundamental physical principles that allow reliable extrapolation of accelerated tests, understanding of degradation mechanisms and design of materials with targeted properties.

5 Synergies with other SET- Plan technologies

Where possible full use should be made of technology developments in other industries where these can benefit the materials and processes in a cost effective manner. An example would be to take advantage of the heat exchanger developments underway in conventional power (fossil fuel) plants and the extensive research programmes into 700 °C steam generators. Such developments may offer better economic alternatives for the near term and feed new ideas to extend existing material limits in the longer term. More advanced alternatives may be provided through synergies with areas such as Fusion technology (dissimilar and similar material bonding, advanced materials selection) which can provide new solutions and opportunities to share the costs of new materials developments across more than one technology field.

Temperature	RT to 1000 °C			RT to 1200 °C		
Radiation	0-200 dpa			0 dpa		
Environments	Water, steam, impure helium, liquid metals, molten salts			Gases (gasification, combustion), steam, water, low melting point eutectics, air		
Typical loading	Gen II	Gen III+/IV	Fusion	Boilers (incl. UHTC)	Steam Turbines	Gas Turbines (Jet/landbased)
Strength	X	X	X	X	X	X
Ductility	X	X	X	X	X	X
Toughness (KIC)	X	X	X	X	X	X
Creep strength	-	X	X	X	X	X
Creep ductility	-	X	X	X	X	X
LCF strength	X	X	X	X	X	X
HCF strength	X	-	-	-	X	X
Corrosion	X	X	X	X	X	X
Microstructural changes	Irradiation induced	Irradiation induced	Irradiation Induced	Thermally induced	Thermally induced	Thermally induced
Subcritical crack growth	X	X	x	X	X	X
Irradiation damage	X	X	X	-	-	-
Creep-fatigue	-	X	X	X	X	X
Fatigue-environment interactions	X	X	x	-	X	-
Creep-fatigue-envir. interactions	-	X	X	X	X	X

Table 5.1 matrix for possible nuclear/non-nuclear interactions

The fusion development requirements for structural materials partly overlap with Generation IV target areas with respect to fast neutron fluence levels and temperature ranges, but the helium (and hydrogen) generation issues provide more extreme demands for fusion. The commonality in ferritic-martensitic ODS development would lie predominantly in the primary production processes, whereas shaping requirements are different: thin-walled tubes versus plates. In both areas adequate fracture properties and creep-fatigue performance are key for reliable operation and high plant availability, and the prospective for high efficiency.

At the same time the back-end issue for large steel volumes of consumables is up-front accounted for in the fusion plant and material design, while the fission spent fuel characteristics are dominated by the fuel's radioactive inventory. In both technology areas materials recycling technologies will have to be developed, with the perspective of economic large scale implementation meeting future societal and market demands. Indications for improved performance and potentially for back-end issues are already evident from the observed performance of "clean steels".

Materials such as advanced nickel based alloys and super-alloys and composite materials have experienced significant development in the Gas Turbine, Aerospace and Aircraft industries and a

transfer of technologies from these areas could reduce the time needed to develop reliable alternatives suitable for Fission applications.

Introduction of new materials technologies is typically component specific, and integrated with qualification and performance monitoring methodologies. Introduction of composite materials e.g. will require a tailored qualification path, which is essentially different from qualifying base materials. Timely codes and standards development and the associated knowledge base is a prerequisite for introduction of any new materials technology in the nuclear industry. Further synergies will be obtained in developing qualification methodologies in extreme environments.

Another example concerns the development of graphite technologies, where high purity demands in substrate technologies will be beneficial for reduced activation of HTR core structures and improve the economic prospects for the back-end technologies within the total life-cycle analysis.

Exchanges with non-nuclear technologies should be positively promoted to ensure effective exchanges and collaborative actions including the establishment of joint conferences and seminars. The research infrastructure with academic parties, institutes, industrial parties and utilities and governmental bodies should be enforced, in order to meet the demanding technological goals and the human resources for the near and longer term. These include physics based and mechanistic modelling computations at all scales, as elaborated more in the next section.

Temperature	RT to 1000 °C			RT to 1200 °C		
Radiation	0-200 dpa			0 dpa		
Environments	Water, steam, impure helium, liquid metals, molten salts			Gases (gasification, combustion), steam, water, low melting point eutectics, air		
Materials	Gen II	Gen III+/IV	Fusion	Boilers (inc.UHTC)	Steam Turbines	Gas Turbines (Jet/landbased)
Low alloy steel	X	X	-	X	X	X
Ferritic/bainitic	-	X	-	X	X	X
Ferritic/martensitic	-	X	X	X	X	X
austenitic	X	X	X	X	X	X
duplex	-	-	-	X	X	-
Superalloys						
Solid solution	X	X	-	X	X	X
Gamma prime	X	X	X	-	-	X
Intermetallics	-	X	-	-	-	X
Nanostructured (ODS, gradient,bulk)	X	X	X	-	X	X
Refractory alloys	-	X	X	-	-	X
Ceramics (C, SiC, Oxides)	-	X	X	-	-	X
Coatings (corrosion, erosion, wear)	-	X	x	X	X	X

Table 5.2 Materials Cross-cut Areas

In addition specific advancements will be obtained in development of metallic, ceramic and polymer sensors, detectors and other instrumentation for safety and control of subsystems condition and coolant chemistry in highly radioactive and/or contaminated environments.

For Europe it should be interesting to build or maintain a domestic high-technology materials industry for a wide range of applications. Nano-tailored materials (ODS, nano-clusters, nano-engineered coatings), advanced ceramic concepts, joining and surface treatments will commercialize once sound market potential becomes apparent. It is therefore most important to establish connections to and synergies with non-nuclear users. Joint materials projects (like EXTREMAT-IP) should be encouraged and supported.

For effective deployment of innovative material technologies, the nuclear industry requires a tailored research and testing infrastructure. This includes representative test environments, like irradiation facilities, also allowing accelerated tests:

1. Preliminary tests on nuclear technology materials properties
2. Generation of design data on interaction of environmental effects
3. Generation of data for design, and design limits.
4. Preliminary subcomponent testing
5. Component testing for design validation.

The cross-cutting with other SET areas following the workshops can be summarised as follows, including EU based industry supply chains:

- Large forgings for various steels
- Advanced welding techniques
- Dissimilar joints ferrite/austenitic/Ni-alloys/ODS-materials
- High temperature performance & creep rupture limits
- Corrosion management, coatings and repair
- Long term life management of components
- Composite design, codes and standards
- Strategic: alloying elements and fibre suppliers

Synergies with other SET-Plan technologies identified at Rapporteur's meeting

After the Rapporteurs meeting with the other SET-Plan technologies currently under consideration have been identified in the following way:

Fossil fuel including CCS

Interaction mainly for structural materials for combustion

- **Common material issues:** *High temperature strength* as e.g. creep; corrosion, oxidation etc.
- **Key materials:** Nickel based steels, F/M steels, ODS steels and thermal barrier coatings. In the long-term composites and ceramics could be of interests.
- **Topics:** R&D for material development and qualification, as well as welding, fabrication routes, NDT qualification.

Direct Solar

- **Common materials issues:** *High temperature strength*; issues concerned with molten salt and liquid metals as heat transfer media (e.g. compatibility).
- **Key materials:** Steels and Protective coatings, qualification of very large structures and heat exchangers.
- **Topics:** R&D for materials development and qualification as well as activities for commercial deployment.

Bio-energy

- The bio-energy shares many issues for structural materials with **fossil fuel**.

Wind energy

- **Topics:** Development of models and methodologies to predict and mitigate material degradation such as stress corrosion cracking. Design rules, damage monitoring and reliability. Another issue is large components and welding technologies.

Fuel cells and hydrogen

- **Topic:** *High temperature strength* for heat exchangers for hydrogen production.

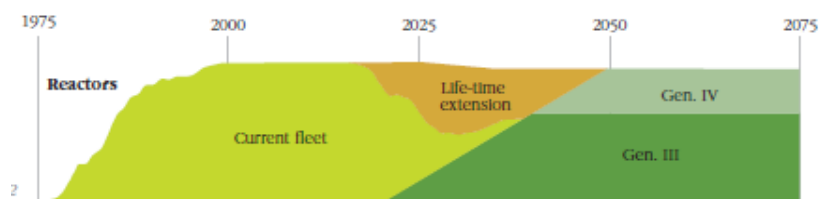
PV, storage, grids

- For PV, storage and grids the link to nuclear fission is weaker.

- The storage rapporteur said that for long-term R&D there could be close links with nuclear energy. The concept of beta-voltaic, which is based on nuclear decay was mentioned.

6 Needs and recommendations of activities addressing 2020 and 2050 market implementation

All the three pillars identified by SNETP give important contributions to the SET-Plan goals and different activities with different time lines are needed for each of the pillars. The two nuclear SET-plan tools: the European Sustainable Nuclear Industrial Initiative and the European Energy Research Alliance (EERA) Nuclear Materials address the three fast reactor concepts, SFR, LFR and GFR and ADS. The needs and recommendations on these reactor concepts are an important part of the activities in this roadmap. The light-water reactors will, however, be the dominant source for nuclear energy even by 2050 but with fast reactors gradually increasing their share of the electricity production as illustrated in Figure 6.1.



6.1: Deployment of Generation III and Generation IV reactors for electricity generation³

The process heat application and co-generation by high temperature reactors also plays an important role in reaching the general SET-Plan goals.

From the technological assessment it has been realised that for all types of reactor technologies, materials development and materials performance assessment requires:

- Dedicated infrastructure that should become available at the European or international level to perform real or realistic tests
- Pre-normative research and standardization of the qualified materials
- The continuation of R&D efforts on multi-scale modelling by development of physical model and tools and validation with well-defined experiments.

To satisfy these needs, Europe will have to develop high quality and up-to-date facilities for both irradiation and testing (MTR, Hotlabs) and post-test analysis, to perform well defined and controlled experiments for materials optimisation and qualification.

6.1 Current LWR technology (Generation II) and LWR of Generation III type

The needs and recommendations for R&D activities are primarily related to the LTO of Generation II LWR, which includes the plant life extension, the power up-rate and possibly high burn-up and high conversion objectives. Moreover, design lifetime objective of Generation III reactors is 60 years and high fuel burn-up as well as high fuel conversion are objectives currently evaluated. Understanding materials and welds degradation mechanism can reduce the uncertainties related to LTO.

Key target and needs for activities are:

- Activities related to the understanding and prediction of ageing and embrittlement mechanisms and consequences of reactor pressure vessels steel (normally low alloy bainitic steel). This can be done by optimizing surveillance programmes (e.g. trend curves for higher fluences, Master Curve applications) as well as by developing adequate models, also based on modelling-oriented experiments, using if relevant model alloys.

³ SNETP Vision report 2007

- Activities related to the understanding and prediction of environmental assisted and irradiation assisted stress corrosion cracking in austenitic steels (used for in-core and structural components) and Ni based alloys.
- Environmental effects on mechanical properties: Effect of the environment on fatigue code curves for stainless steels, low temperature crack propagation of nickel-based weld metals
- Synergies of thermal and neutron embrittlement in cast stainless steels
- Fuel assembly materials performance: in this case attention is put on Zr alloys, Ag/In/Cd alloys, B4Cm austenitic steel and Ni alloys. The main concerns to be addressed are possible in-pile deformations.

The suggested activities are related to:

- Accelerated experimental tests: Materials are tested under pre-defined and well controlled conditions to determine either the behaviour of the material or a specific parameter of this same material. These tests should always be accurate (high quality of data), reproducible (statistically relevant) and different time duration and with some very long duration. These tests constitute the basis for the prediction of the long-term material behaviour and need to be supported by advanced microstructural assessment tools.
- A major problem with assessing degradation mechanisms is that there is lack of in-pile material that has been subjected to actual environmental conditions. The decommissioning of older reactors could therefore provide a unique opportunity for accessing 'in operation in-pile material'. There are also components from surveillance programmes, some of them have been in the reactors for several decades. We therefore propose an activity dedicated to the collection of such data and experimental programme for assessment for long-term degradation mechanisms.
- Fundamental understanding of the possible mechanisms causing degradation or failure of a component has a tremendous importance as it is the only way to validate the relevance of accelerated tests for real conditions. Thus, the actual effort of the EU *building the tools for the multi-scale modelling approach* should be pursued and much more collaboration between different disciplines should be re-enforced and strengthened.

The recent very unfortunate Fukushima accident indicates that among others, some of the issues related to the materials degradation have to be looked at in detail, more specifically (not exhaustive)

- primary system corrosion and chemistry:
 - o corrosion of shafts, valve/disks, etc. from "dirty water"
 - o contaminated water processing and storage/disposition
- enhanced equipment /system performance
- Steam/air oxidation of Zr-alloys at high temperatures:
 - o In-depth analysis and understanding to mitigate this issue
 - o Development of advanced fuel cladding material to avoid Zirconium/steam interaction at 1200°C which produces Hydrogen gas (coatings, new materials).
- development of reliable instrumentation and definition the adequate parameters to monitor in case of severe accident.

In this context the development of advanced fuel cladding material to avoid Zirconium/steam interaction at high temperature, thus avoiding hydrogen build – up and explosion suggest the development of new materials. The research on new materials is at a very early stage and several alternatives can be investigated as e.g. Silicon Carbide matrixes or composites or as coating. Other options might be considered as well.

6.2 New Technology Generation IV / ADS and ESNII systems

The current strategy for implementation of Generation IV and ADS reactor systems as defined by the SNETP is to develop and build for the short to medium term (2020/2030) prototypes and demonstrator and to perform R&D activities for the long-term milestone which foresees the industrial implementation. The medium term activities have been identified within the ESNII initiatives. The

Strategic Research Agenda of the SNETP outlines a schedule for materials development, which includes key milestones and supporting activities. This schedule is illustrated in Figure 6.2.

The implementation of new materials in reactor systems needs several steps before materials can be used for nuclear components. These steps include:

- Screening for materials specification,
- Materials specification, fabrication manufacturing, welding and testing,
- Qualification and codification,
- Validation for all conceivable conditions for the specific component.

The very long-term perspective due to the very rigorous safety requirements is a characteristic of nuclear materials research and development.

In Appendix 1 and the paragraphs below, materials and needed activities for the ESNII prototype and demonstrator, and materials and needed activities for the industrial scale systems are addressed. Moreover paragraphs are dedicated to the pre-normative research (essential task for all materials that are developed for nuclear components) and to the physics based models / tools development for materials prediction.

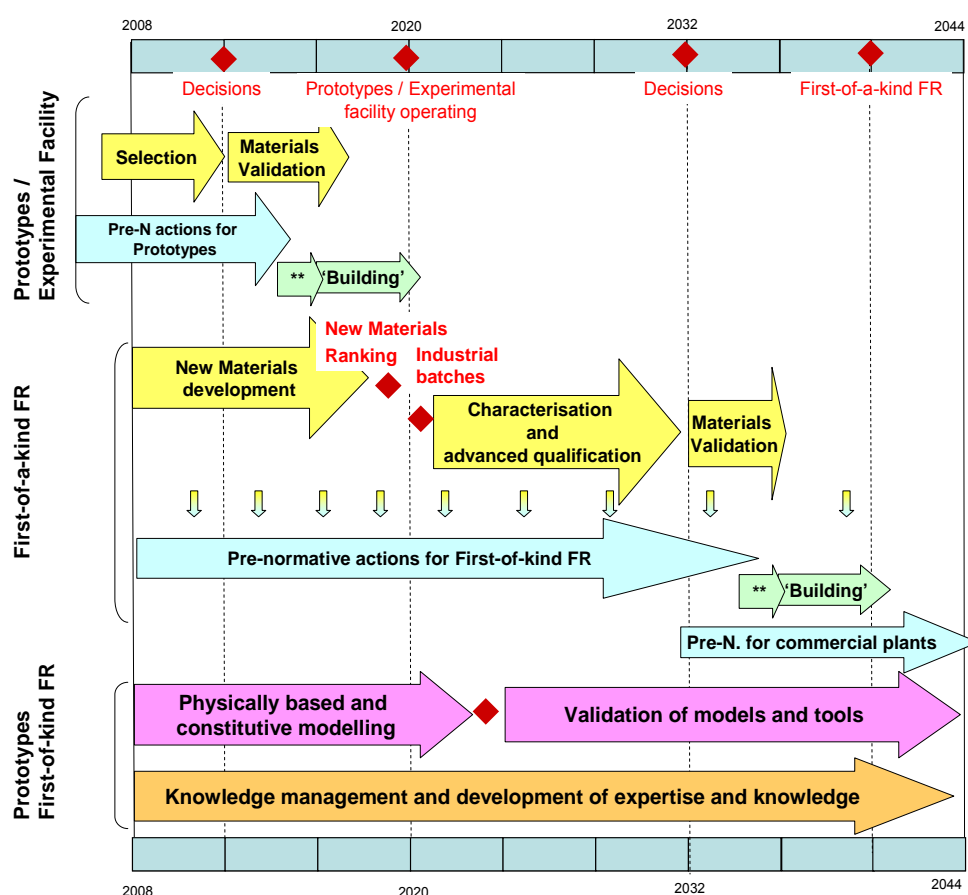


Fig 6.2: Materials development schedule for Generation IV and ESNII systems as defined discussed within the SRA of SNETP

6.3 Nuclear Cogeneration and (V)HTR technologies

The current strategy for the development of nuclear co-generation includes high temperature reactor systems as well as small sized LWR's with lower quality heat. It is to be noted that at the time of compiling this report the Nuclear Cogeneration to Industry (NC2I) initiative is actually being launched. An integrated list of needs and recommendations for the 2020/30 and 2050 market implementation is not yet (June 2011) available from this new industrial initiative.

The strategy and activities for the cogeneration pillar of SNETP have been identified in the Strategic Research Agenda. The key systems and major materials issues for HTR & (V)HTR have been elaborated in chapters 3 and 4. The implementation requires deployment of mainly existing materials

in the 2020/30 timeframe and new materials in the 2050 timeframe. In addition there are many close synergies between materials and component needs between the HTR and GFR system (see Appendix 2), particularly in the heat transport circuit where for both systems the main components will be largely the same. The VHTR system operates at potentially higher temperatures than the GFR and therefore acts as a pilot for many of the issues that also need to be addressed by GFR. Except for the core, much of the R & D work on materials and components performed on the (V)HTR is therefore of direct benefit to the GFR system. In the 2030 perspective and an outlet temperature of up to 800 °C.

- Qualification of the materials and components needed for the construction and licensing for the deployment of commercial co-generation reactors. This involves ferritic-martensitic steels, nickel-based alloys, composites and deployment of gas to gas and gas to steam heat exchange systems.
- Qualification of graphite as core material for helium cooled HTR at intermediate temperatures (up to 800° C)

Looking at the 2050 perspective and commercial deployment of high temperature process heat applications (above 850 °C)

- Qualification of the materials needed for the construction and licensing for the deployment of higher powered commercial innovative reactors. This involves ferritic-martensitic steels, nickel-based alloys, composites and ceramics.
- Qualification of graphite for VHTR with $T > 950^{\circ}\text{C}$.

6.4 Physics-based models, Model experiments and validation

It parallel to the experimental investigations, a large effort should also be directed towards physically-based models that address features and processes for different length and time scale and that can provide physical insight. This includes molecular dynamics, dislocation dynamics, crystal plasticity but also continuum mechanics. The appropriate model and length scale depends on the specific mechanism and feature. In many cases understanding material properties, their degradation and interaction with the coolants involves several length and time scales. This means that we need to use several scales, and methodologies to pass the information between the scales is fundamental. Modelling activities are of relevance for all classes of reactors and for all classes of materials development since fundamental understanding of phenomena helps in defining mitigation strategies, optimisation of materials specification and optimisation of experimental programs.

Perhaps the most important application of physics-based models is that it could allow us to understand the behaviour under extreme conditions and long-term, for which we have very little or no experimental data. Important examples are extrapolation to end-of-life conditions and transferability from laboratory scale to component. The physics based models are also central in understanding the various degradation mechanisms. They could also be helpful tools in material and component development since different designs and parameter combinations can be assessed and processes simulated.

The goals in a short-to-medium and in a long-term perspective are:

- In a short-to-medium term perspective, *design-oriented modelling* (DOMO) activities are needed. DOMO activities will combine semi-empirical and physics-based models, together with suitable modelling-oriented experiments involving in-depth materials characterisation, to provide background information and/or tools usable by materials scientists and designers for selecting materials and conditions for specific reactor concepts and basic understanding of degradation mechanisms. These models should be applied for the 2020/30 time frame.
- In a long-term perspective, *physics-based modelling* (PBMO) activities will need to be launched. The ultimate goal is to integrate the different models into multiscale common platforms, in the timeframe perspective of 2050. PBMO activities will make use of the most advanced calculation and computer simulation techniques including refined experiments, having as first and foremost the goal of a full understanding of the mechanisms governing the behaviour of materials under the extreme conditions expected in future (but also present) reactors and their inclusion in suitable physical models.

In short, DOMO activities should provide an answer to a specific question in a shorter-term perspective, while the underlying motivation of PBMO activities is the development of sophisticated physics-based models, to be eventually integrated in a multi-scale scheme.

The effective theory construction, taking advantage of the increasing (but always limited) computational power, more sophisticated microstructural techniques and experimental data will be the main activity. This applies to the models at specific scales as well as the information passage between the scales towards integrated multi-scale models.

6.5 Pre-normative research and Design Codes

Pre-normative research should exploit the more basic research into tools standardisation, procedures and design codes. For Generation IV power plants, the status and the roadmap of the different projects naturally lead to separate the pre-normative activities into three steps:

- The short term issues (2012) with pre-normative actions focusing on the tools for design and construction of ESNII relevant and other systems facilities based on existing data
- The medium term issues (2020) deal with the R&D results to answer the technical challenges for the ESNII and other systems,
- The long term issues (2040) aims to consolidate feedback from prototypes and from the development of commercial power plants.

R&D challenges for the short, medium and long term

Design and Construction Codes provide a set of essential engineering tools for the design assessment and construction of systems components. They define the common reference between prime contractors, operators, designers, engineers, manufacturers, suppliers, inspectors and safety authorities. They define the quality level of equipment necessary to meet nuclear standards. Whenever new materials are used or loading and environmental conditions change, the design codes need to be modified and extended accordingly. Pre-normative research is required to advance existing codes and standards.

It is essential to reinforce European cooperation on the development of new nuclear system equipment for the Generation IV reactors. This can be done through pre-normative actions whose main objectives would be to capitalize R&D results on materials, structural behaviour analysis, joining, welding, fabrication and non-destructive examinations, to bring together best European practice and harmonize criteria, codes and test procedures.

A sound European basis for these objectives is RCC-MRx, which is based on the feedback from the design and construction of Superphenix (RCC-MR) and the Jules Horowitz material test reactor (RCC-Mx).

Procedures for material and component testing is another important pre-normative research area. Important aspects include miniature specimen tests and transferability between laboratory conditions and long term operation.

Short term issues:

Although the industrial deployment of commercial Generation IV nuclear systems is planned for the long-term, first operation of ESNII systems is planned by ~2020. These very near milestones require that evaluation of different technological solutions be completed within the next few years.

With these milestones, the short term pre-normative priorities should focus on the rules for design and construction of the ESNII facilities, namely:

- mechanical properties;
- fabrication processes;
- identification of potential damaging phenomena for new materials / new coolant systems;
- review and critical analysis of the current RCC-MR version;
- R&D focusing on design rules for very high temperature conditions.

- Assessment of design rules for defect tolerance, and associated inspection requirements;

These actions should be based on the development of existing codes such as RCC-MRx code to provide design assessment and construction rules in time for the technology assessment within the next few years and the following construction contract discussion phase. In parallel we need a road-map for a European Design Code that addresses the different innovative reactor systems should be discussed with European stakeholders, including safety experts and fusion research society. This has already been initiated under the auspice of CEN/CELEC, based on RCC-MRx and complemented with experience and feedback from other codes such as the UK R5 and R6 assessment procedures. The basic RCC-MRx design assumes linear elastic conditions with extension to non-linear material and fracture mechanics in Appendices. A major effort is probably needed to address coolant compatibility for lead. With increasing temperatures there will also be a need to address non-linear effects more.

Medium term issues (2020):

New components and materials as well as new combination of materials / coolants are investigated. For these options pre-normative actions are needed in terms of design rules, materials, fabrication (including joining technologies), NDE techniques and key material properties for the specific environment.

Another domain of investigation is the design of components under high dose irradiation or with significant creep deformation, where the interaction of creep and other damage mechanisms remain an open question.

The industrial deployment of Generation IV reactors worldwide also calls for harmonizing Design and Construction Codes. A harmonized international code in terms of design and construction codes ought to be defined and developed, particularly with safety experts, stakeholders as e.g. those present within the Generation IV International Forum Senior Industry Advisory Panel.

Long-term issues (2040):

Feedback from ESNII and other reactor components will necessarily lead to new development in the different domains covered by the design and construction codes:

- Completion of material specifications with the support of manufacturers
- Update of codification rules for manufacturing, welding and examination processes
- Design rules would probably have to take into account new domain of working and eventually new degradations.

In parallel, research work for new materials need to be maintained and material properties tabulated on a database system.

6.6 Materials R&D needs for the Industrial systems

6.6.1 Commercially available materials; 9Cr F/M steel, austenitic steels and Ni-based superalloys

A number of materials that exist today have been identified as candidate materials in several of the innovative reactor concepts. These materials have not been used at large scale for the more severe conditions or for the specific components. To this end a research programme needs to be performed during the design phase. This includes:

- Characterisation and validation of already available materials
- Development of testing procedures to account for environmental effects on mechanical properties
- Joining and welding procedures (including mixed welding, welding of thick components etc.)
- Creation of materials database including coolant (primary and secondary), thermal, mechanical and irradiation effects (single and combined effect)
- Integrity assessment of components lasting for long-time operation.

Even if several classes of materials have been identified for the various components of the four ESNII systems, it turned out that three classes of materials, namely the austenitic steels AISI 316L and Ti stabilised 15Cr-15Ni, the modified ferritic/martensitic 9Cr steel and Ni alloys have been considered for use in more than one ESNII system and also for HTR and SCWR. The research areas above should be addressed for these materials by taking into account commonalities and specific items components and reactors for which the materials applications have been envisaged.

Pilot Project: 9Cr steel heat exchanger:

Phase 1 Manufacturing: The selection of 9Cr steel as heat exchanger material for the innovative system is driven by the better thermal conductivity of this material with respect to austenitic steel and the lower thermal expansion. Indeed, for the liquid metal systems this class of steel is indicated as a one option to be considered for the building of the heat exchanger (with primary and secondary coolant). Materials data generated in the focus area 1 will allow the full characterisation of this material and will drive the design of the heat exchanger where also welding and joining are essential items to be tested.

Phase 2: Technology Testing: Heat exchanger testing should be performed in appropriate facilities to demonstrate the feasibility of the concepts with the selected material and manufacturing / welding procedures. The tests should be performed with the relevant heat transfer media and in relevant thermal and mechanical conditions. The test parameters should account for normal operational conditions as well as identified transients.

6.6.2 ODS Materials

Many challenges must be met before the nuclear industry will qualify and use these ODS materials. These range from the elaboration of the material through its shaping, welding, effect of environment, behavior under irradiation. The essential steps are described below together with the R&D needs.

Selection of materials and elaboration processes

Currently, ODS materials are obtained by powder metallurgy. The different production steps include powder production, milling and consolidation that can be achieved either by hot extrusion or by Hot Isostatic Pressing (HIP). In some manufacture routes, two steps are chosen with an initial consolidation by HIP followed by hot extrusion. To produce tubes it is necessary to use the hot extrusion to obtain the mother tubes. These steps are followed by a sequence of cold rolling and heat treatments to stress relieve the material.

The process described above is the industrial one which was used to produce few tens of tons of ODS per year with nano-oxides. Other processes are under study around the world.

The optimization or the development of new elaboration processes concern all the referred applications. The mechanical alloying followed by consolidation is the standard fabrication route and different optimization of the process can be proposed. It includes the parameters of the mechanical alloying (duration, atmosphere, addition elements) or the consolidation process and the transformation route.

On the other hand other fabrication routes have to be assessed. For example, ODS could be produced by injecting nano yttrium-oxide, via a side nozzle, into droplets of molten steel. Once the droplets are doped, they solidify to form an ingot which can be transformed. Other techniques are also studied by EB-PVD (Electron beam physical vapor deposition).

Depending on the application, the chemical composition of the ODS has to be optimized. For instance, in case of fuel cladding for SFR, the Fuel-Cladding-Chemical-Interaction and the behaviour during the reprocessing are essential requirements for ODS composition selection. For lead or LBE cooled reactors or Gas Reactors the compatibility is a further essential item to be considered for ODS composition selection. In all cases and for the different environments, an evaluation of corrosion or degradation mechanisms needs to be identified as well as modeling of the corrosion processes to guarantee the life time of the component.

The welding/joining of ODS is a critical point. During the operation, the microstructure has to be maintained as unchanged as possible. Processes without fusion will be therefore favored.

The transition or the transfer between lab heats and industrial quality production is a major concern as well and specific attention and characterizations will be necessary.

Basic topics for materials under operation / Mechanical properties, environment and irradiation impacts

Different topics need to be considered, for example the mechanical behaviour (softening phenomena, embrittlement under irradiation....) or the evolution of the microstructure (understanding of phase stability under irradiation, trapping of defects at ODS particles). Those activities will be conducted using the data obtained on irradiated and non-irradiated ODS but also on model alloys and from analytic experiments like ion irradiation to simulate the neutron damage.

Depending on the application and the operation conditions, the ODS materials will sustain different temperatures, mechanical loadings, environment and irradiation conditions. The qualification of these materials implies the need to design and perform specific experiments, in particular under irradiation. The possibility to irradiate optimized ODS at relevant doses will depend on the availability of the experimental and industrial reactors. Only a few reactors can be considered all over the world for this task. Thus, the new irradiations will be designed with great care and the treatment of previous irradiations with ODS will be finalized in order to obtain the maximum amount of data and feedback. As the number of integral experiments will be limited, it is necessary to develop modeling tools to understand, model and extrapolate the results and to guarantee that the behavior of the material will be satisfactory in all the conditions.

Cross cutting topics: tests and data

The micro-structural and mechanical characterization of ODS materials responds to very specific problems. The materials are nano-reinforced and nano-structured. For the microstructure it implies characterization at the nano-scale with particular tools like high resolution microscopy, Tomographic Atom Probe, high resolution diffraction. Unlike many metallic materials, the final mechanical properties of the component depend strongly on its shape and on its fabrication process. It implies the need for the development of a specific testing procedure, which will be standardized, for each major component. For example, standard experiments do not exist to evaluate the Ductile to Brittle Transition Temperature of ODS tubes in all the directions or rules to determine or estimate the toughness of the ODS tubes. The data management for ODS materials is also a specific challenge mainly because of the very particular relationships between the microstructure and the mechanical behavior.

Summary list of R&D needs

- Selection of alloy composition appropriate for specific applications via screening
- Impact of manufacturing routes on properties (including economy and quality)
- From scale lab heats to an industrial quality production
- Innovative materials either reinforced by other nanoparticles or based on other processes of fabrication
- Welding/joining: Stability of microstructure during heat-treatment
- Mechanical properties properties: softening phenomena, Low temperature embrittlement of ductility and fracture toughness at low temperature, Creep and fatigue resistance, Qualification of anisotropy (if it occurs) and impact on properties
- Stability of ODS particles under long term ageing and under irradiation, trapping of defects at ODS particles, effect of ODS particles on irradiation damage in matrix material.
- Thermal and irradiation ageing of ODS steel below 500°C
- Improvement of corrosion resistance at high temperatures (in liquid metals)
- Fuel cladding chemical interaction (coupling between the mechanical behaviour of the fuel and the cladding and the atmosphere – tellure- fission gas)
- Irradiation behaviour at low temperatures as function of Cr content
- common procedures for material testing
- Collect and manage materials data

Key performance indicator:

- Availability of ODS cladding (optimised composition and industrial manufacturing procedure) able to withstand (including dimensional stability):
 - temperatures of 400-650°C,
 - irradiation doses higher than 150 dpa
 - creep life exceeding 80 000 hours for 100 MPa,
 - Coolant compatibility and fuel/cladding mechanical and chemical interaction.
 - Suitability for reprocessing.

Pilot Project on ODS:

Phase 1 Manufacturing: Cladding tubes with appropriate composition as required for the liquid metal systems should be produced with the selected production route in representative dimension. Appropriate joining technologies should be available to close the tubes with their end-caps. Moreover, depending from the design of the reactor core spacers should be applicable to the cladding. The entire pin as well as pin bundles should be made available for full scale testing.

Phase 2: Technology Testing: The tests should be aimed at demonstrating the concept, which should include also properties related to the entire fuel cycle. The test matrix of the single pin and the fuel bundle should include normal operational conditions and predicted transients. The ODS cladding should also be tested in the relevant environments, e.g. Na or Pb.

6.6.3 SiC/SiC

SiC/SiC has been envisaged as fuel cladding for GFR and is also a future candidate material as fuel cladding in high temperature LFR. SiC/SiC has also been proposed for several internal subcomponents and control rods in (V)HTR. Moreover, the interest of SiC/SiC is also available from the fusion technology community. Essential activities for SiC/SiC development include:

- structural composite manufacturing development,
- establishing new standard test methods,
- dedicated testing infrastructures,
- characterization program oriented to the development of data for design and also for modelling activities and investigation of environmental effects.

Structural composite manufacturing program (manufacturers, materials experts, design experts)

The main complication of composites is that they are uniquely engineered for their specific application and thus no off-the-shelf component is commercially available. Each component geometry and technical requirement will dictate the best fibre architecture design, aided by stress analysis codes which may require improvements if 3D complex reinforcement by braiding/weaving is needed.

The main large-scale European manufacturers of composites: Snecma in France, Eads, MT Aerospace, AG, SGL and Schunk in Germany, are not qualifying their high temperature performance composite manufacturing and processing routes. Composites are currently identified by manufacturer, composition and other characteristics. The development of formal materials and process specification along with rigorous process control and monitoring are essential requisites for bringing candidate composites to a more advanced level (prime candidate), such that an appropriate characterization program can be made to qualify them for structural application in nuclear environment. Prime candidate production process must be examined for reproducibility.

Current manufacturing capabilities may present practical limitations to the size and shape of component that can be realized. Scaling-up of the parts may require significant investment. Scaling-up may also implicate different infiltration process efficiency and may require process adjustment. Moreover, an increase in the capacity for producing the appropriate interface coating on the reinforcing fibres may be necessary as well as other fabrication processes. Moreover, improvement concerning actual joining/coating technologies and development/approval of qualification methods are also needed.

Establishing new standard test methods

Grade specific design data covering the full range of design data such as tensile, compressive, flexural and shear strength, thermal expansion and conductivity, fracture toughness and oxidation resistance are needed. There is a need for standardisation in quality assurance (QA) and test method pertaining to the acquisition of these data. A number of test standards for ceramic composites have been developed by the relevant Technical Committees of the existing National/International Association for Standardization: ASTM (C28-07), CEN (TC184-SC1), ISO (TC206), AFNOR (B43-C). Unfortunately, despite this level of development, current Standards are useful for comparative purposes (screening tests) but likely inadequate for design, due to the fact that composites are considered to be engineered material systems for their specific end-use. As such, the raw materials, composite architectures as well as the processing methods affect the properties of the final product. Thus new standards must be developed for non-standard shapes of the different components, specially engineered to optimize their performances relative to the anticipated loads and environmental conditions. Test specimens with different size and geometry should be evaluated to establish if they are truly representative of the full-scale component.

Characterization program

The characterisation program foresees the short term tests that should include mechanical tests for design basic-data (normal/off-normal) and other property tests. Examples are

- stress-strain tensile curves (Young modulus, proportional limit, ultimate tensile strength);
- fracture properties (fracture toughness, strength of fibres etc)
- low-cycle fatigue properties;
- Other physical property tests (coefficient of thermal expansion, thermal conductivity, and specific heat).

Moreover Long-term tests (time-dependant phenomena) include:

- Creep, high cycle fatigue and creep-fatigue interaction tests
- Microstructural evaluations → input data for multi-scale modelling.
- Development of phenomenological material constitutive models (continuous damage accumulation) for the inelastic design
- Non-linear and time dependent tensile and compressive properties;
- Development of phenomenological failure models for design criteria;
- From continuous damage model to more physically-based discrete micro damage model including He effects.

Environmental tests comprise:

- Corrosion tests (including off-normal chemistry)
- Long-term ageing tests (normal chemistry)
- Microstructural observation.
- Kinetics (rate) laws.
- Mechanical stresses / Environment combined effect

Irradiation tests

- Effect of temperature/dose
- PIE: dimensional stability and physical properties, mechanical tests (short) and microstructural evaluations → input data for multi-scale modelling.
- Irradiation creep
- Mechanical / Environment and Irradiation combined effect

Pilot Project SiC_s/SiC for fuel cladding:

Phase 1 Manufacturing: Cladding tubes for the gas cooled system (and possibly also for Pb cooled system) with appropriate and qualified production route as outlined above should be made available. The tubes should have representative dimensions and end-caps should be welded / joined with qualified technologies. Single pin and pin bundles should be made available for a full scale testing of this concept.

Phase 2: Technology Testing: The tests should be aimed at demonstrating the concept. The testing approach should include single pin and pin bundle tests. The testing parameters should account for

normal operational conditions and identified transients. The cladding SiC_fSiC should be tested in the relevant environments being this He and Pb out of pile and when ESNII system available also in-pile.

Key performance indicator:

- Availability of SiC_fSiC cladding (optimised) able to withstand:
 - nominal operational temperature of 900-1100 °C;
 - irradiation dose > 60-80 dpa (in SiC);
 - mechanical stresses from coolant and internal stresses due to fuel swelling, fission gas release and thermal through-thickness temperature gradients; and be:
 - compatible with coolant (He, Pb) and with carbide fuel;
 - suitable for reprocessing.

6.6.4 Coatings

Coatings are considered as important items to be developed for corrosion protection purposes or thermal barriers for different reactor concepts. In the case of HLM cooled systems FeAlCrY coatings are considered at present reference systems. However, other coating materials are also under investigation. For R&D needs on coatings, the key items to be addressed are:

- definition of reference composition for the specific application;
- optimisation of fabrication technique;
- definition of testing rules for qualification;
- characterisation in relevant environment and under all conceivable conditions.

Pilot project Coatings:

Phase 1 Manufacturing: Coated cladding tubes with selected reference composition and selected coating manufacturing technology. The welding of end-caps should be integrated in the component development as well as the need of spacers. The requirements are dictated by the Pb and Pb-Bi and Na systems developed.

Phase 2 Technology Testing: The tests should be aimed at demonstrating the concept. The testing approach should include single pin and pin bundle tests. The testing parameters should account for normal operational conditions and identified transients. The coated claddings should be tested in the relevant environments for example: Pb and Pb-Bi out of pile and when ESNII system available also in-pile.

Key performance indicator:

- Availability of
 - high quality fuel cladding coating and qualified coating procedure.
 - the coating should be adherent and coherent, and
 - it should be able to withstand all conditions dictated by the cladding itself.
 - The coating should possibly show self-healing properties.

6.6.5 Other materials

In the case of LFR Ti₃SiC₂ (MAXTHAL) has been identified as one of the most promising candidate material for the pump impeller.

The preliminary results gained during recent investigations suggest further investigations of the class of Titanium Aluminides materials for fast and high temperature reactors. TiAl has shown some interesting properties for future nuclear applications. Because of their superior creep strength they could provide an alternative to solution strengthened superalloys. The material should be further investigated. Irradiation behaviour of and helium effects in TiAl should be studied with globular second phases. Production technologies cast or powder metallurgy need to be established.

6.7 Infrastructures

An infrastructure is indispensable for the implementation of R&D programme and construction and licensing of prototypes and demonstrators. The project ADRIANA has made an overview needed

facilities. These needs will be met by refurbishment of existing facilities and construction of new ones. Key infrastructures for materials research include:

Irradiation facilities: Irradiation facilities are needed for the development of nuclear materials. Jules Horowitz in France, now under construction will provide neutron flux for Gen III and generic Gen IV research, MYRRHA will provide fast neutron spectra to support Gen IV research and PALLAS in Petten is proposed to replace the high flux reactor.

Hot laboratories: A number of hot-laboratories equipped with advanced measurement and testing techniques are mandatory for the development of materials.

High Temperature testing systems: High temperature testing systems to assess mechanical properties (tensile, creep, fatigue, etc.) in relevant environments are mandatory for the screening and characterisation of materials.

Large scale facilities for out of pile testing of components: Out of pile testing of developed components can be performed in large scale facilities that operate in representative conditions. Several facilities are already available, however it should be checked if additional ones are needed (e.g. for transient testing) for a complete characterisation programme.

The costs to build and operate such facilities are quite high. It is therefore necessary that this is shared by the European member states.

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Appendix 1 Detailed requirements for the ESNII prototypes

A1.1 Sodium Fast Reactor Prototype ASTRID

Austenitic steels for core or structural components

The first materials that will be used for many components in Astrid are austenitic steels type 316L(N). These materials will experience high temperature and/or creep or creep-fatigue (vessels, inner structures, heat exchanger, circuits...). Their chemical specification was adjusted during SuperPhénix conception. The feedback from their use in the past is satisfying. But the key issue is the integrity of 316L(N) components in service conditions **for 60 years at least**. For such a long time, modelling and eventually extrapolation is needed, as well as long term laboratory tests.

Two key material issues for base metal and welded joints need to be addressed:

- the reliability of the extrapolation method (either with models or with a simple engineer assessment), **deeper understanding of basic degradation mechanisms**
- the need of **long term results** for creep, creep-fatigue, corrosion in Na, thermal ageing in air, limited irradiation submission...

Other core materials

The design of the new core for Astrid implies an increase of the diameter of the cladding tubes as well as a decrease of the amount of sodium inside the core. The reference material for the cladding of the first assemblies is the austenitic steel: **15/15 Ti AIM1**. This material is qualified for the Phénix and Super Phénix conception but **complementary data and optimisations concerning the fabrication route for example need to be conducted on this material for the Astrid reactor**.

In addition, the **Fe-9CrMo martensitic steels** are considered for the wrapper tube. Once again, the geometry and the requirements for the Astrid wrapper tubes are slightly different compared to the ones for the previous Sodium Fast Reactors. Different optimisations need to be conducted on this type of materials. The **mixed welding** between Fe-9Cr martensitic steels and austenitic materials (i. e. 316) has to be addressed as the joint between these two types of materials are necessary for sub-assemblies.

9%Cr steels for steam generators

At present time, the use of this material is possible for steam generators in Astrid. They have interesting thermal characteristics and the international community tends to trust this material (IGCAR, JAEA...). But some actions are still necessary to fully decide of their relevance.

Several key material issues (~priority order):

- mechanical behaviour: to quantify cyclic softening in service conditions, to evaluate microstructural stability and impact resistance after long term thermal ageing, to deal with low ductility at high temperature (forming, respect of ESPN rules...)
- compatibility with environments: 9Cr steels are sensitive to water / steam oxidation ; limited long term corrosion in sodium needs to be checked
- welded joints: their creep and ageing behaviour needs to be checked
- in service conditions, 9Cr steels may not be as resistant as some other steels to certain risks: stress corrosion cracking in soda zones, wastage and resistance to water / sodium reaction propagation, hydrogen embrittlement at high temperature.

Resistant alloy coatings

Cobalt alloy coatings (e.g. stellite) are currently used in pressurized water reactors as hard and wear resistant surfaces in taps and valves, or for protection of internal structural components in sodium fast reactors. Due to their own well-known activation by neutron irradiation, those alloys are responsible for primary circuit activation. This long standing issue has not been solved yet and it is necessary to monitor the R&D activities for replacement of stellites and to propose some solutions for Astrid.

In this context **screening tests** are needed to find the best candidates, referring to specifications in terms of corrosion in sodium, thermal cycling resistance, thermal ageing and tribology / friction forces.

Other structural materials

Alloy 800 or 800H, if 9Cr steels are proved to be not relevant for steam generator applications.

Alloy 800 was used in steam generator of SuperPhénix and is also being considered for HTR. If this alloy is chosen for Astrid, there is a need to update the knowledge and industrial knowhow for manufacturing thermal exchange tubes and to integrate in RCC-MX the specifications, mechanical characteristics and welding procedures for this alloy. Microstructural and mechanical characteristics depend on the product that will be chosen. Past studies need to be referred to and summarized, in order to identify future needs for base metal and welded joints (ageing, creep and creep-fatigue, environments and resistance to risks in service...)

304(L) for some cast pieces

These steels were largely used in Phenix and SuperPhenix; there is a need to update the knowledge and codification requirements (manufacturing, chemistry, welding, mechanical characteristics...).

Low alloyed steels for elements that will not be submitted to creep solicitation (lower in-service temperature)

These steels were also used in Phenix and SuperPhenix, there is a need to update the knowledge and codification requirements (manufacturing, chemistry, welding, mechanical characteristics...).

A1.2 MYRRHA and Lead Fast Reactor Demonstrator ALFRED

MYRRHA

MYRRHA is an Accelerator Driven System (ADS) under development at the SCK•CEN Mol in Belgium and is serving as a basis for the European XT-ADS (eXperimental demonstration of Transmutation in ADS) to provide protons and neutrons for various R&D applications and to be used as materials test reactors (such as the BR2).

One of the critical challenges of the MYRRHA structural materials is their compatibility (in terms of corrosion and mechanical resistance) with the coolant, namely liquid metal Lead Bismuth Eutectic, under intense neutron irradiation (see Table A-1).

The properties of the candidate materials for MYRRHA that needs to be investigated are corrosion rate and mechanical properties as tensile, fracture toughness, fatigue, creep. Moreover Environmental assisted cracking, which can be the underlying cause for a series of materials properties degradation effects, should be thoroughly investigated.

Because of the limited experience in the compatibility of structural materials an extensive R&D program is needed to provide reliable data to support the materials selection. The experimental programs should include irradiation of structural materials in LBE and their further characterization under various conditions.

Although MYRRHA is conceptually designed to an advanced level, the final choice for structural materials is not yet fully decided, Table A-1 reports potential candidate materials.

A number of components (structural materials close to the spallation target (cladding, diaphragm, core support plate) in MYRRHA will be exposed to intense neutron flux in presence of LBE and complex thermo-mechanical loading. For these components it is relevant to verify both their ***dimensional stability and their structural integrity***.

An extensive experimental program in the un-irradiated condition is necessary to ***identify the critical parameters and provide testing and evaluation guidelines that will be systematically used in the future***. At present, there are no standards for the kind of tests that are performed worldwide and also with regard to specimen size and configuration, specimen preparation, testing conditions (loading rate for example), control of the environment, result evaluation and interpretation differ from laboratory to laboratory.

In this sense all experiments proposed for LBE and Pb cooled systems require significant effort to ***generate a representative database of experimental results*** and to ***implement these data as rules and materials properties in design codes***.

ALFRED

As for MYRRHA also for ALFRED the major technical issue is related to the structural material compatibility in liquid Pb and under intense neutron flux. To minimise the risk related to this issue, a low temperature thermal cycle has been proposed with 400°C core inlet temperature to have sufficient margin above the lead freezing point and only 480°C mean core outlet temperature, with many advantages for the structural steels in term of corrosion, creep, and reduced thermal shocks in transient conditions. Table A-2 shows the materials selected for the various components.

As far as the fuel cladding is concerned, this can be protected against corrosion by surface aluminization or by the use of functionally graded composites for a greater safety margin.

As far as the materials characterisation program is concerned, similar requirements and needs as expressed for MYRRHA can also be envisaged here.

For the long term LFR concepts, where the reactor coolant outlet temperature is expected to be increased up to 700-800°C, the development of new materials, able to withstand both high dpa and high coolant temperatures, is required. Therefore, while the development of fuel claddings operating in pure lead environment under fast neutron flux irradiation, in the temperature range 400°C–600°C is needed in the short term, the characterization of ODS steels, SiCSiC composites and “MAX phase” materials is required for increasing the reactor operating temperature, enabling higher efficiency in energy generation as well as hydrogen production. In parallel, special attention has to be paid to the development of advanced coatings for steel protection. Specific needs related to these materials are reported in the paragraph 1.6.6.

Components	Replacability	Coolant	Tmax	Dpa dose	Candidate material
Beam window	Yes	LBE Inlet: 300°C	500°C (Tentative)	40dpa/... 5-15 appm He/dpa	T91 (or)
Fuel cladding	Yes	LBE Inlet: 300°C Outlet: 450°C	500°C		15-15Ti (or T91)
In reactor components (Core barrel etc.)	Yes/No	LBE Inlet: 300°C Outlet: 410°C	400°C		T91 (or)
Reactor vessel	No	LBE Inlet: 300°C Outlet: 410°C	300°C		(316L) (.....)

Table A-1: MYRRHA candidate structural materials.

Component/Subcomponent	Material	Replaceability (Y=yes, N=no)
Diagrid	AISI316L	Y/N
Fuel Assemblies	T91	Y
Dummy Assemblies	T91	Y
Core Restrainer Plate	AISI316L	Y/N
Pump duct upstream the pump	AISI316L	Y/N
Primary Pump (<i>Shaft, Impeller, Casing</i>)	MAXTHAL (Ti ₃ SiC ₂)	Y
Pump duct downstream the pump	AISI316L	Y
Steam Generators	T91	Y
Reactor Vessel	AISI316L	N
Inner Vessel	AISI316L	Y/N
Decay Heat Removal Heat Exchangers (Dip Coolers)	T91	Y
Purification System	AISI316L	Y

A1.3 Gas Cooled Fast Reactor Demonstrator ALLEGRO

ALLEGRO will be the first step towards the electricity generating prototype GFR. The main goals of this demonstrator are:

- to demonstrate the viability of the GFR reactor system and to establish confidence in the innovative GFR technology
- to qualify specific GFR technologies such as the fuel, the fuel elements (U, Pu ceramic carbide or nitride with SiC/SiC_f cladding and wrappers), and specific safety systems, in particular, the decay heat removal function
- branches on the main intermediate heat exchanger will allow the testing and validation of high temperature components and processes

The ALLEGRO demonstrator layout represents a loop-type non-electricity generating reactor where the secondary coolant is pressurised water and the final heat sink is the atmosphere to avoid the use of high temperature materials.

The core design includes a two-step approach:

- the first core (start-up core) using conventional MOX fuel and steel cladding with some experimental GFR fuel sub-assemblies; the core outlet temperature will be limited to 560 °C
- the second core (demonstration core) using only GFR reference fuel (carbide fuel with ceramic cladding); the core outlet temperature will be as high as 850 °C

The key parameters of the ALLEGRO reactor are given in Table 3 for both the start-up and the demonstration core.

For what concerns the primary circuit and internal structures requirements, the following items must be considered

- *Bottom plate and core support* are structures located below the core and operate at the core inlet temperature (400 °C in the case of the demonstration core)
- *Core barrel liner* is a structure surrounding the reactor core and the reflector structure and operates at temperatures up to 500 °C
- *Hot gas duct* carries hot He from the core to the heat removal system; it faces high temperature of 850 °C on one side (the demonstration core) and low temperature of 400 °C on the other side
- *Cross vessel* carries cold He toward the core operates at temperature of 400 °C
- *Upper plenum shroud* operates at high temperature up to 850 °C

Modified 9Cr-1Mo steel

9Cr – 1 Mo steel is a good candidate for **all metallic out-of-core structures**. The out-of-core components will be subjected to low radiation damage << 1 dpa calculated for 60 years life. All internal metallic structures are exposed to cold He (~400 °C) on one side and if necessary are protected by thermal shields from the hot He (850 °C) on the other side. Furthermore, all internal structures will be subject to mechanical stresses in the range of 10 to 80 MPa and also to thermal stresses due to thermal cycling. The material must be also compatible with He impurities.

The reactor pressure vessel (RPV) will consist of a cylindrical shell and hemispherical bottom. It will have 3 m in diameter, with a height of 8 m and the wall thickness of 100 mm. Total mass of the RPV including the head will be around 120 tons. The RPV has to withstand the pressure of 7 MPa at 400 °C in normal operating conditions and at ~550 °C in reference off-normal transients. Modified **9Cr-1Mo steel** appears to be the best candidate material for ALLEGRO RPV since characterizations of 9Cr1Mo steels are needed for base metal and welds samples in the appropriate thickness range. The main lack is on fabrication capacities for vessels. Through synergies between GFR and HTR this subject has been addressed in the development of the HTR and within the Raphael project.

Thermal-shielding materials

As far as the thermal shielding is concerned, conventional VHTR thermal barriers are proposed to be used in ALLEGRO. An example of such barriers are Al₂O₃ and SiO₂ mixed ceramic fiber materials

contained between metallic or carbon-carbon cover plates attached to the primary structures that require insulation. In addition, these barriers can be covered with a shield made of carbide or SiC felts or foams. These thermal barriers must withstand He velocities of about 60 ms^{-1} and depressurisation rates in the range of 2 MPas^{-1} . This needs the development of specific GFR solutions and their qualification on relevant facilities. The development roadmap includes:

- Development of high temperature material, like carbide or SiC felts or foams,
- Design of thermal barrier structure,
- Experimental qualification, with the manufacture of a new device which allows the test of thermal barriers in accidental GFR conditions.

Data on the manufacture and performance of the thermal insulation systems are needed to ensure that the selected materials are capable of lasting for the life of the reactor. The data includes: physical properties (heat conductivity, heat capacity), long term thermal and compositional stability, mechanical strength at temperature, resistance to pressure drop, vibrations, corrosion resistance to moisture- and air-helium mixtures, resistance to irradiation.

Ni-alloys

The heat exchanger in the ALLEGRO system will be only a cooler (heat transfer from hot He to water) to avoid problems with high temperature materials. However, there will be a possibility to install an additional high temperature 10 MW gas-to-gas heat exchanger to test the high temperature components. The intermediate heat exchanger (IHX) will be the most critical out-of-core component of the GFR reactor.

The thermo-mechanical environment ($850 \text{ }^{\circ}\text{C}$ primary He, secondary He/Nitrogen or He/Ar) induces the use of **Ni-base alloys** such as **HR230** or **Inc617**. Large uncertainties on the component lifetime remain due to the geometric complexity and the lack of material properties including welds. Qualifications tests of the component at reduced scale will be necessary. This component is not critical for ALLEGRO which only needs coolers (no energy conversion) but it is essential for the GFR. Again because of synergies between the GFR and HTR this subject has been addressed in the development of the HTR within the Raphael project.

Structural materials of in-core applications.

	Primary circuit		Secondary circuit
	<i>Start-up core config.</i>	<i>Demonstration core config.</i>	
Coolant	Helium		Water
Helium pressure (MPa)	7.0		6.5
Core power (MWth)	75	75	-
IHX max temperature ($^{\circ}\text{C}$)	533	810	197
IHX min temperature ($^{\circ}\text{C}$)	260	400	130
Core inlet/outlet temperature ($^{\circ}\text{C}$)	260/560	400/850	
Fuel/fuel assembly	(U,Pu) O_2 / pin	(U,Pu)C / pin	
Max. cladding/fuel temperature ($^{\circ}\text{C}$)	615/1045	1045/1140	
Power density (MW/m^3)	100-150	100-150	

Table A-3. The key parameters of the ALLEGRO reactor

The start-up core: AIM1

The start-up subassemblies (S/A) will be based on pin-type MOX fuel elements with metallic cladding and metallic wrapper material. The maximum fuel temperature will be $1045 \text{ }^{\circ}\text{C}$ and the maximum temperature of the cladding will be $615 \text{ }^{\circ}\text{C}$. The maximum level of damage will be 24 dpa Fe.

The **candidate material** for the start-up core fuel element **cladding** and **wrapper** tube is **AIM1 austenitic stainless steel**, ODS steel and vanadium alloys are the alternatives.

The demonstration core: SiC/SiCf

The reference fuel element geometry for the demonstration core S/A is at the moment the pin whereas the plate-type honeycomb structure is a back-up alternative. The target criteria for the cladding are:

- Clad temperature in normal conditions of 1000°C,
- No fission product release for a clad temperature of 1600°C during a few hours,
- Maintaining the core cooling ability up to a clad temperature of 2000°C.

The maximum level of damage will be 70-80 dpa SiC.

The **candidate material** for the demonstration core fuel element **cladding** is **SiC/SiCf** ceramic composite. The candidate material for S/A wrapper is also SiC/SiCf.

Appendix 2 Synergies between (V)HTR and GFR for Co-generation

There are many close synergies between materials and component needs between the HTR and ENSII GFR system particularly in the heat transport circuit where for both systems the main components will be largely the same. The VHTR system operates at potentially higher temperatures than the GFR and therefore acts as a pilot for many of the issues that also need to be addressed by GFR. Much of the R & D work on materials and components performed on the (V)HTR is therefore of direct benefit to the GFR system. Areas of synergy are as follows:

Commonality of component requirements of HTRs with the GFR

- High temperature heat exchangers:
 - Main HX, He to He-N₂ – normally small primary to secondary pressure differential, T~850°C, Power ~ 800 MWth / unit
 - DHR HX, He to water, Power ~ 50 MWth / unit
- Steam Generators – The same unit can be used for both HTR and GFR for process heat and electricity generation.
- Gas circulators – He environment, 5~10 MW, submerged magnetic bearings
- He purification plant
- He circuit valves, power operated and passive check valves (probably check valves with a powered override)
- High temperature coaxial pipes (straight and curved)
- Control rod drives
- In-reactor fuel handling equipment
- Reactor pressure vessels and core barrel and in-vessel insulation systems.
- Equipment for process heat off-take.

Commonality of material requirements of HTRs and the GFR:

- Pressure vessel and core barrel materials – hot vessel concept:
 - Mod 9Cr 1Mo steel non-cooled vessel
 - Core Barrel and core support materials where inlet gas temperature are the same
- Co-axial ductwork materials
- SiCf reinforced SiC for fuel cladding and fuel element wrapper tubes
- SiC main heat exchanger tubes for very high temperatures
- Refractory metals (vanadium and molybdenum) as back-up materials to ceramics.
- ODS as the “back-up” back-up material (for a moderate temperature GFR operating with steam plant or SC-CO₂)
- Thermal insulation materials
- Surface treatments to eliminate tribology problems in He

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Abstract

This scientific assessment serves as the basis for a materials research roadmap for the nuclear fission technology, itself an integral element of an overall "Materials Roadmap Enabling Low Carbon Technologies", a Commission Staff Working Document published in December 2011. The Materials Roadmap aims at contributing to strategic decisions on materials research funding at European and Member State levels and is aligned with the priorities of the Strategic Energy Technology Plan (SET-Plan). It is intended to serve as a guide for developing specific research and development activities in the field of materials for energy applications over the next 10 years.

This report provides an in-depth analysis of the state-of-the-art and future challenges for energy technology-related materials and the needs for research activities to support the development of nuclear fission technology both for the 2020 and the 2050 market horizons.

It has been produced by independent and renowned European materials scientists and energy technology experts, drawn from academia, research institutes and industry, under the coordination the SET-Plan Information System (SETIS), which is managed by the Joint Research Centre (JRC) of the European Commission. The contents were presented and discussed at a dedicated hearing in which a wide pool of stakeholders participated, including representatives of the relevant technology platforms, industry associations and the Joint Programmes of the European Energy Research Associations.

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